UNITED STATES NUCLEAR REGULATORY COMMISSION

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

MASSACHUSETTS INSTITUTE OF TECHNOLOGY REACTOR

RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-020

License No. R-37

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for renewal of Facility Operating License No. R-37 filed by the Massachusetts Institute of Technology (MIT, the licensee) dated July 8, 1999, as supplemented by letters dated February 10, and May 8, 2000, January 29, 2004, July 5, and October 11, 2006, January 26, 2007, February 22, May 29, August 15, August 21, August 26, October 6, October 7, and December 1, 2008, May 26, August 27, October 5, October 9, and November 19, 2009, and March 30, August 6, and August 26, 2010, (the application), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10, Chapter 1, of the *Code of Federal Regulations* (10 CFR);

   B. Construction of the Massachusetts Institute of Technology reactor facility (the facility) was completed in substantial conformity with Construction Permits Nos. CPRR-5 and CPRR-118, 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 that (i) the activities authorized by this renewed license can be conducted at the designated location without endangering the health and safety of the public, and (ii) 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 license in accordance with the rules and regulations of the Commission;

F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements,” of the Commission’s regulations;

G. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

H. The issuance of this renewed license is in accordance with 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” of the Commission’s regulations and all applicable requirements; and

I. The receipt, possession and use of byproduct and special nuclear materials as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.”

2. Facility Operating License No. R-37 is hereby renewed in its entirety to read as follows:

A. This license applies to the Massachusetts Institute of Technology reactor facility that is owned by the Massachusetts Institute of Technology, located on MIT’s campus in Cambridge, Massachusetts, and described in the licensee’s application, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses MIT:

1. Pursuant to subsections 104a and 104c of the Act, and 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” to possess, use, and operate the facility as a utilization facility at the designated location in accordance with the procedures and limitations described in the application and this renewed license.

2. Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use in connection with operation of the facility:

a. up to 40 kilograms of contained uranium-235 of any enrichment, including up to 100 grams for use in fueled experiments, provided that not more than 1.6 kilograms of this amount be unirradiated;

b. two 1-curie plutonium-beryllium neutron sources; and

c. such special nuclear material as may be produced by operation of the facility, which shall not be separated.

3. Pursuant to the Act and 10 CFR Part 30, to receive, possess, and use:

    a. a 150-curie antimony-beryllium sealed neutron source in connection with operation of the facility;

    b. such byproduct material as may be produced by operation of the facility, which, except for byproduct material produced in non-fueled experiments, shall not be separated; and

    c. byproduct materials activated in reactors other than the MIT reactor (for use in the reactor hot cells) that are in solid form and have atomic numbers 3 through 83. The total inventory of this byproduct material shall not exceed 100,000 curies at any one time. This material may be irradiated in the reactor.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in Parts 20, “Standards for Protection against Radiation,” 30, 50, 51, 55, “Operators’ Licenses,” 70, and 73, “Physical Protection of Plants and Materials,” of the Commission’s regulations; is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

Maximum Power Level

1. The licensee is authorized to operate the reactor at steady-state power levels not to exceed 6.0 megawatts (thermal).

Technical Specifications

2. The Technical Specifications contained in Appendix A, as revised through Amendment 46, are hereby incorporated in the license. The Massachusetts Institute of Technology shall operate the facility in accordance with the Technical Specifications.

Additional Conditions

3. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The approved physical security plan consists of a Massachusetts Institute of Technology Nuclear Reactor Laboratory document, withheld from public disclosure pursuant to 10 CFR 73.21, entitled, “Physical Security Plan for the M.I.T. Research Reactor Facility,” dated July 22, 2013, as revised.

D. This renewed license is effective as of the date of issuance and shall expire at midnight twenty years from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A, Technical Specifications

Date of Issuance: November 1, 2010

TECHNICAL SPECIFICATIONS

FOR THE MIT RESEARCH REACTOR (MITR-II)

(Rev. 6)

NUCLEAR REACTOR LABORATORY
MASSACHUSETTS INSTITUTE OF TECHNOLOGY
CAMBRIDGE, MASSACHUSETTS

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1. INTRODUCTION

These technical specifications apply to the MIT Research Reactor, which is designated as the MITR-II, and to its associated experimental facilities.

1.1 Scope

The following areas are addressed: Definitions, Safety Limits (SL), Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements, Design Features, Experiments, and Administrative Controls.

1.2 Application

1.2.1 Purpose

These specifications are derived from the MITR-II's Safety Analysis Report (SAR). They consist of specific limitations and equipment requirements for the safe operation of the reactor and for dealing with abnormal situations. These specifications represent a comprehensive envelope of safe operation. Only those operational parameters and equipment requirements directly related to preserving this safety envelope are listed.

1.2.2 Format

The format of these specifications is as indicated in Section 1.2.2 of ANSI/ANS-15.1-2007.

1.3 Definitions

1.3.1 Channel

A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

1.3.2 Channel Calibration

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

1.3.3 Channel Check

A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same parameter.

1.3.4 Channel Test

A channel test is the introduction of a signal into the channel for verification that it is operable.

1.3.5 Containment

Containment is an enclosure of the facility designed to (1) be at a negative internal pressure to ensure in-leakage, (2) control the release of effluents to the environment, and (3) mitigate the consequences of certain analyzed accidents or events.

1.3.6 Containment Integrity

Integrity of the containment enclosure (building) is said to be maintained when all isolation system equipment is either operable or secured in an isolating position.

1.3.7 Damaged Fuel

The term "damaged fuel" means: (1) a failure to meet fuel fabrication specifications unless the deviation was jointly pre-approved by MIT and U.S. Department of Energy Quality Assurance based on satisfaction of the MITR-II SAR, or (2) a deterioration of the clad is present that results in fission product levels associated with an element that are elevated by a factor of five or more above the average background level for the core as a whole. The increase in level is measured in the immediate vicinity of the element in question when the reactor is shutdown.

1.3.8 Excess Reactivity

Excess reactivity is the amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is exactly critical (K-effective = 1.0) at reference core conditions.

1.3.9 Experiment

An experiment is any operation, hardware, or target (excluding devices such as detectors, foils, etc.), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the core tank, or in a beamport or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be a part of their design.

1.3.10 Experimental Facility

An experimental facility is an appurtenance to the reactor that is generally used to contain and orient an experiment, as in the case of an irradiation thimble, or to provide a desired flux distribution, as in the case of a filtered beam.

1.3.11 Frequency

Each required surveillance test or other function shall be performed within the specified time interval with:

a) A maximum allowable extension not to exceed 25% of the specified surveillance interval, unless otherwise stated in these Technical Specifications.

b) A total maximum combined interval time for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

c) Where indicated in the surveillance Technical Specifications, scheduled surveillances that cannot be performed while the reactor is shut down may be deferred until the next planned period of reactor operation. Such surveillances shall be performed as soon as practicable when reactor operation resumes.

Surveillance tests required for experiments (Section 6) may be waived when an instrument, component, or system is not required to be operable, but any such instrument, component, or system shall be tested prior to being used as a required operable instrument, component, or system.

1.3.12 Immediate

Immediate means that the required action will be initiated without delay in an orderly manner by using written procedures when applicable.

1.3.13 Inadmissible Sample Materials

Those materials defined by the MIT Reactor Safeguards Committee (MITRSC) as either not allowable within the MITR-II or restricted from the reactor containment building. Examples include unapproved amounts of combustible, corrosive, or explosive materials.

1.3.14 Independent Experiments

Experiments that are not connected by a mechanical, chemical, or electrical link.

1.3.15 Irradiation

Use of reactor experimental facilities where the primary purpose is the production of activated material such as samples for neutron activation analysis, or materials that exhibit radiation damage effects, or radioactive isotopes, or other similar activities. An irradiation may also refer to use of a medical therapy irradiation room for human therapy or other activities such as radiography.

1.3.16 Irradiation Series

A series of irradiations reviewed simultaneously on the basis of the maximum material, flux, and irradiation time of any sample of the series.

1.3.17 Measured Value

The measured value is the value of a parameter as it appears on the output of a channel.

1.3.18 Movable Experiment

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

1.3.19 Non-Secured Experiment

A non-secured experiment is one where it is intended that the experiment should not move while the reactor is operating, but the experiment is held in place with less restraint than a secured experiment.

1.3.20 Operable

Operable means a component or system is capable of performing its intended function.

1.3.21 Operating

Operating means a component or system is performing its intended function.

1.3.22 Potential Reactivity Worth of an Experiment

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.

1.3.23 Protective Action

A protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit.

1.3.24 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

1.3.25 Reactor Operating

The MITR-II is operating whenever it is not in either a secured or a shutdown condition.

1.3.26 Reactor Operator

A reactor operator is an individual who is licensed by the U. S. Nuclear Regulatory Commission to manipulate the controls of the MIT Research Reactor.

1.3.27 Reactor Safety System

The MITR-II's safety system consists of those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. The MITR-II reactor safety system is also referred to as the reactor protection system.

1.3.28 Reactor Secured

The MITR-II is secured when:

1. Either there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection, or
2. The following conditions exist:
    a) The minimum number of neutron absorbing control devices are fully inserted or other safety devices are in a shutdown position, as required by technical specifications,
    b) The console key switch is in the off position and the key is removed from the lock,
    c) No work is in progress involving core fuel, core structure, installed control devices, or control device drives unless they are physically decoupled from the control devices,
    d) No in-core experiments are being moved or serviced, and
    e) No work is in progress involving fuel in the fission converter tank.

1.3.29 Reactor Shutdown

The MITR-II is shut down when all control devices (shim blades and regulating rod) are fully inserted or a reactivity condition exists that is equivalent to one where all control devices are fully inserted.

1.3.30 Reference Core Condition

The reference core condition is the reactivity condition of the core when the primary coolant and D_2O reflector coolant temperatures are at 10° C and the reactivity worth of xenon is zero (i.e., cold, clean, and critical).

1.3.31 Regulating Rod

The MITR-II's regulating rod is a low worth control device that is used primarily to maintain an intended power level. It does not have a scram capability. Its position may be varied manually or by an automatic controller.

1.3.32 Reportable Occurrence

A reportable occurrence is any of the following:

1. Any actual safety system setting less conservative than specified in the MITR-II Technical Specifications except during periods of instrument maintenance with the reactor shut down,

2. Operation in violation of a limiting condition for operation,

3. Safety system component malfunction or other component or system malfunction which renders, or which threatens to render, the safety system incapable of performing its intended function,

4. Release of fission products from a fuel element in a quantity that would indicate a fuel element cladding failure,

5. An uncontrolled or unanticipated change in reactivity greater than 0.78% $\Delta K/K$ (1 beta),

6. An abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks),

7. 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,

8. Conditions arising from natural or offsite manmade events that affect or threaten to affect the safe operation of the facility.

Refer to Specification 7.7.2 for additional reporting requirements.

1.3.33 Research Reactor

The term research reactor as used in these Technical Specifications refers to the Massachusetts Institute of Technology's Research Reactor which is licensed by the U.S. Nuclear Regulatory Commission under license No. R-37. It supports a self-sustaining neutron chain reaction for research, developmental, educational, training, medical, and experimental purposes. It is also used for the production of non-fissile radionuclides for use in medical treatments and other purposes and for the medical treatment of humans using neutron beams.

1.3.34 Review and Approve

The terminology "shall review and approve" is to be interpreted as requiring that the reviewing group or person shall carry out a review of the matter in question and may then either approve or disapprove it. Before it can be implemented, the matter in question must receive an approval from the reviewing group or person.

1.3.35 Safety Analysis Report

The Safety Analysis Report (SAR) is the document submitted to the U.S. Nuclear Regulatory Commission on July 8, 1999, entitled, "Safety Analysis Report for the MIT Research Reactor (MITR-II)," and subsequent revisions thereof.

1.3.36 Safety Channel

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

1.3.37 Scram Time

The scram time, or shim blade insertion time, is the time elapsed between the initiation of a scram signal and movement of the shim blade from its current position to its 80% inserted position.

1.3.38 Secured Experiment

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

1.3.39 Secured Shutdown

Secured shutdown is achieved when the reactor meets the requirements of the definition of "reactor secured" and the facility administrative requirements for leaving the facility with no licensed reactor operators present.

1.3.40 <u>Senior Reactor Operator</u>

A senior reactor operator is an individual who is licensed by the U. S. Nuclear Regulatory Commission to direct the activities of reactor operators at the MIT Research Reactor. Such an individual is also a reactor operator.

1.3.41 <u>Shall, Should, and May</u>

The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

1.3.42 <u>Shim Blade</u>

MITR-II shim blades (also called control rods) are devices fabricated from neutron-absorbing materials that are used to establish neutron flux changes and to compensate for routine reactivity losses. The MITR-II shim blades are coupled to their drives by electromagnets and they perform a safety function when the electromagnet is de-energized. Shim blade position may be varied manually or by an automatic controller.

1.3.43 <u>Shutdown Margin</u>

Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition. It shall be assumed that the most reactive shim blade and the regulating rod are fully withdrawn and that the reactor will remain subcritical without further operator action. For the MITR-II, the minimum shutdown reactivity is 1% $\Delta K/K$ and the most restrictive operating condition is cold (10 ˚C), xenon-free, with all movable and non-secured experiments in their most reactive state.

1.3.44 Shutdown Reactivity

Shutdown reactivity is the value of the reactivity of the reactor with all control devices in their least reactive positions (e.g., inserted). The value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

1.3.45 True Value

The true value is the actual value of a parameter.

1.3.46 Unscheduled Shutdown

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or check-out operations.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance. These variables are:

P = total reactor power,
W_p = reactor primary coolant total flow rate,
T_{out} = reactor primary coolant outlet temperature, and
H = height of coolant above top of fuel plates.

This specification also applies to the cladding temperature.

Objective

To establish limits within which the integrity of the fuel clad is maintained.

Specification

1. For forced convection, except as noted in Specification 3 below, the point determined by the true values of P, W_p, and T_{out} shall not be above the line given in Figure 2.1-1 corresponding to the coolant height, H.

2. For natural convection, except as noted in Specification 3 below, the true values for P and H shall be as follows:

Variable	**Safety Limit**
Power, P	250 kW (maximum)
Coolant Height, H	6 feet above top of fuel plates (minimum)

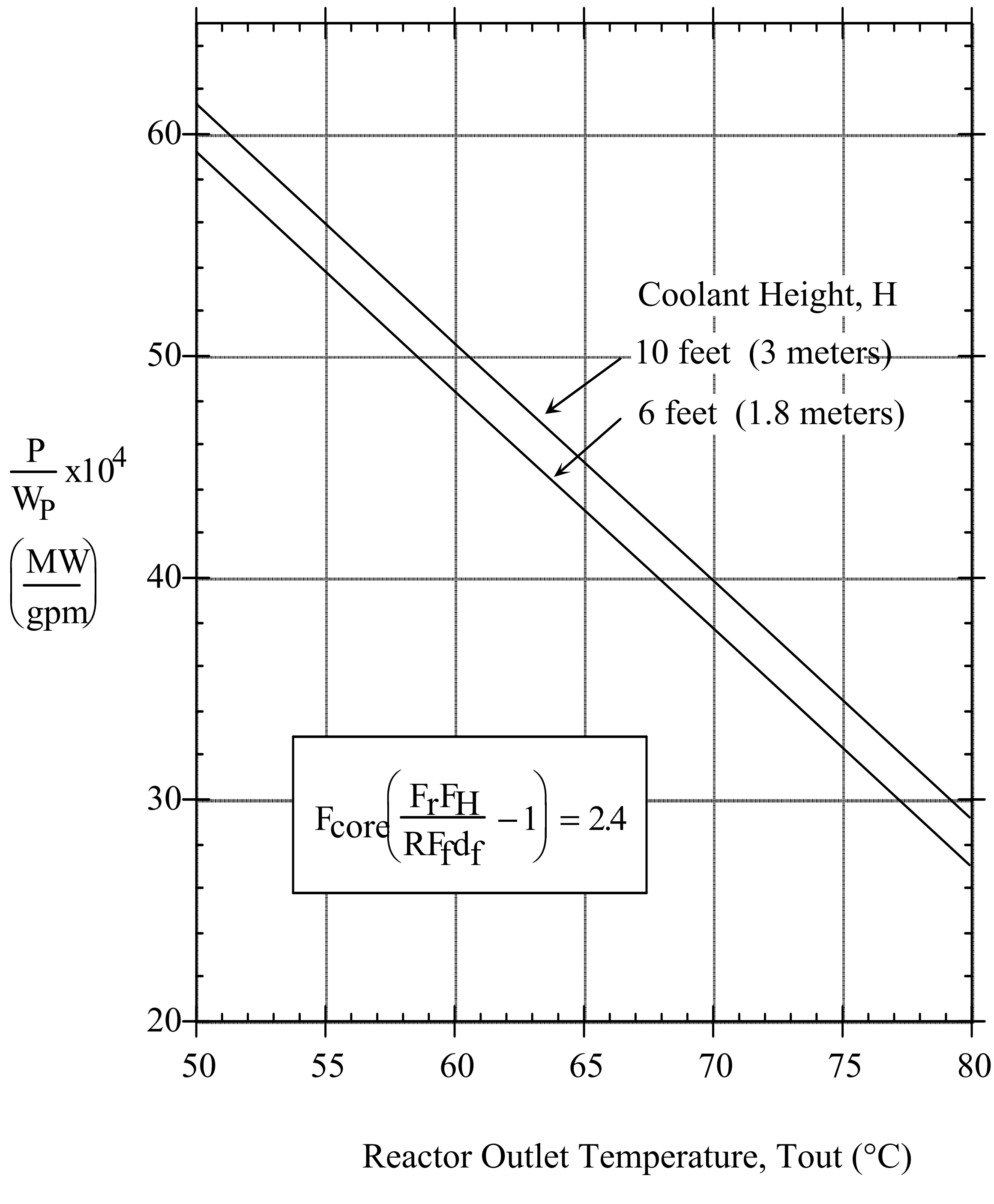

Figure 2.1-1 MITR-II Safety Limits for Forced Convection Operation

3. For transients described in Technical Specification No. 3.1.3, the fuel cladding temperature shall not exceed the cladding softening temperature (450 °C).

Basis

The basis of this specification is given in Section 4.6.5 of the SAR where it is noted that critical heat flux (CHF) is normally used as the criterion of fuel overheating. However, because the coolant flow path in the MITR-II core is a multichannel design, there exists the possibility that flow instabilities could occur before reaching CHF limitations. If onset of flow instability (OFI) did occur first, it would have the effect of lowering the flow rate to the hot channel significantly and thus lowering the critical heat flux. In the safety limit calculations, both CHF and OFI are calculated and the one that would occur first is used to determine the safety limits. Also, in the SAR, a relationship between the reactor operating parameters and OFI is derived with the assumption that the hot channel receives the minimum flow among all the coolant channels. The derivation, which uses the energy conservation equation for the hot channel and the channel subcooling ratio for onset of flow instability, yielded the following relation:

$$\frac{P}{W_p} = \frac{c_{pf}\left(T_{sat} - T_{out}\right)}{F_{core}\left(\dfrac{F_r F_H}{R F_f d_f} - 1\right)} \qquad (2.1\text{-}1)$$

where

F_{core} is the fraction of the total power deposited in the core region,

F_r is the nuclear hot channel factor,

F_H is the engineering hot channel factor for enthalpy rise,

R is the channel outlet subcooling ratio,

F_f is the fraction of primary flow cooling the fuel,

d_f is the flow disparity, which is the ratio of the minimum expected flow in the hot channel to the average channel flow,

c_{pf} is the specific heat of the fluid,

T_{sat} is the water saturation temperature at the outlet end of the core (°C), and

T_{out} is the average core outlet temperature (°C).

The denominator of Equation (2.1-1) is defined as the safety limit factor and is calculated for the MITR-II by assuming R=0.86 (as derived in Section 4.6.2.2 of the SAR), F_r =2.0, $F_f d_f$ = 0.8, F_H =1.173, and F_{core} =1.0. Hence,

$$F_{core}\left(\frac{F_r F_H}{R F_f d_f} - 1\right) = 2.4$$

Figure 2.1-1 shows the safety limits for coolant heights of 10 and 6 feet. The coolant height is the elevation from the top of the fuel plates to the air/water interface at the top of the core tank. A coolant height of 10 feet corresponds to 4 inches below the overflow level. A coolant height of 6 feet corresponds to a point several inches below the anti-siphon valves.

The safety limit factor will be calculated before reactor operation above 1 kW as required by Specification 3.1.4.4, to ensure the validity of the calculated safety limits.

The safety limits for natural-convection operation are calculated using a zero flow critical heat flux correlation. Coolant channels can be cooled by countercurrent flow with a downward movement of water and an upward flow of bubbles or steam generated in the channel. This is referred to as a flooding condition because the flow channels are submerged in a pool of coolant. A detailed description of the zero flow critical heat flux correlation is given in Section 4.6.6.3 of the SAR. For the geometry of the MITR-II fuel elements, the calculated zero flow critical heat flux is 2.353 $\times 10^4$ W/m^2 s, which corresponds to a reactor power of 468 kW with a radial peaking factor of 2.0. Upon taking into account the engineering hot channel factor for enthalpy rise (F_H), the reactor power corresponding to a dryout condition becomes 399 kW. A reactor power of 250 kW is conservatively adopted as the safety limit using a minimum critical heat flux ratio of 1.5. The core outlet temperature and the coolant height do not affect the dryout limit, as long as the core is covered with coolant. The coolant height is conservatively set at 6 feet above top of the fuel plates to ensure an adequate coolant inventory.

The safety limit for transients described in Technical Specification No. 3.1.3, “Maximum Safe Step Reactivity Addition” is the fuel cladding softening temperature (450 °C). For this type of transient, calculational models are used to determine the total energy produced and the associated temperature rise of the fuel. For a 2.3 beta (1.8% $\Delta K/K$) reactivity insertion over period of 0.5 s, the maximum fuel temperature is 83.4 °C [1].

References

2.1-1. File memo dated 9 October 2009.

2.2 Limiting Safety System Settings (LSSS)

Applicability

This specification applies to the setpoints for the safety channels that monitor reactor power, primary coolant flow, reactor outlet temperature, and coolant height above the top of the fuel plates.

Objective

To ensure that automatic protective actions will prevent incipient boiling in the reactor core and will prevent operating conditions from exceeding a safety limit.

Specification

The measured values of the limiting safety system settings on reactor thermal power, P, reactor primary coolant flow rate, W_p, the height of water above the top of the fuel plates, H, and the reactor outlet temperature, T_{out}, shall be as follows:

Table 2.2-1

Limiting Safety System Settings

Parameter	LSSS (2 pumps)	LSSS (1 pump)	LSSS (0 pump)
Power	7.4 MW (max)	3.2 MW (max)	100 kW (max)
Primary Coolant Flow	1800 gpm (min)	900 gpm (min)	N/A
Steady-State Core Outlet Temperature	60 ˚C (max)	60 ˚C (max)	60 ˚C (max)
Coolant Height	4" below overflow or 10 feet above top of fuel plates (min)	4" below overflow or 10 feet above top of fuel plates (min)	4" below overflow or 10 feet above top of fuel plates (min)

Basis

The basis of this specification is given in Section 4.6.7 of the SAR. Onset of nucleate boiling (ONB), which is also called incipient boiling, defines the condition where bubbles first start to form on a heated surface. Because most of the liquid is still subcooled, the bubbles do not detach but grow and collapse while attached to the wall. It is desirable to establish reactor operating conditions that will prevent onset of nucleate boiling because doing so will ensure that a safety limit is not exceeded. In Section 4.6.7 of the SAR, an expression for the ONB limits was derived based on a coolant height corresponding to a pool level 4 inches below the overflow or 10 feet from the top of the fuel plates. This coolant height corresponds to a saturation temperature of 107 ˚C. The engineering hot channel factors for enthalpy rise (F_H) and film temperature rise ($F_{\Delta T}$) are included to account for uncertainties because of measurement, calculation, and possible deviations from nominal design specifications that may affect the thermal hydraulic calculation results.

Forced Convection

The following equation is derived for a coolant channel with an arbitrary axial power deposition distribution q"(z) where z is the axial distance along a fuel plate:

$$T_{out} < 107 + \frac{P \cdot F_{core}}{W_p c_{pf}} + 0.0177\left(q''(z)\right)^{0.466} - \frac{F_H}{\dot{m} c_{pf}} \int_0^z P_H q''(z) dz - F_{\Delta T} \frac{q''(z)}{h} \qquad (2.2\text{-}1)$$

A comparison of three power distribution profiles (uniform, sine/cosine, and bottom peak) was reported in Section 4.6.6.1 of the SAR. This comparison indicated that the uniform power distribution would lead to a maximum clad temperature at the channel outlet. Therefore, Equation (2.2-1) can be simplified to:

$$T_{out} < 107 + \frac{PF_{core}}{W_p c_{pf}} + 0.0177 {q''_{avg}}^{0.466} - \frac{F_H F_r P}{\dot{m} c_{pf} N_c} - F_{\Delta T} \frac{q''_{avg}}{h} \qquad (2.2\text{-}2)$$

where

$$q''_{avg} = \frac{P}{N_c A_H} F_{fuel} F_{core} F_r$$

and

$$\dot{m} = \frac{W_p}{N_c} F_f d_f$$

where

T_{out}	is the bulk outlet temperature in ˚C,
P	is the reactor power,
P_H	is the heated perimeter of a coolant channel,
F_{core}	is the fraction of the total power deposited in the reactor core,
W_p	is the primary flow rate,
c_{pf}	is the heat capacity of the primary coolant,
F_H	is the engineering hot channel factor for enthalpy rise,
$F_{\Delta T}$	is the engineering hot channel factor for film temperature rise,
h	is the heat transfer coefficient,
F_r	is the radial power peaking factor, which is the ratio of the power produced in the fuel plate to the power produced in the average fuel plate,
N_c	is the number of coolant channels,
A_H	is the effective heat transfer area of a fuel plate,
F_{fuel}	is the fraction of the core power deposited in the fuel plates,
d_f	is the flow disparity in the hot channel, and
F_f	is the fraction of primary flow cooling the fuel elements.

The LSSS core outlet temperature as a function of reactor power is calculated using Equation (2.2-2) subject to the assumption of a primary flow rate (W_p) of 1800 gpm (two-pump) and 900 gpm (one-pump) and a coolant height at 4 inches below overflow. Other parameters, such as F_r, F_f, and d_f are the same as specified in the basis of Specification 2.1. Figures 2.2-1 and 2.2-2 show the LSSS reactor power as a function of the core outlet coolant temperature for two-pump and one-pump operation, respectively. For a core outlet coolant temperature of 60 ˚C, the LSSS power is 7.4 MW for two-pump operation and 3.2 MW for one-pump operation.

Natural Convection

Natural convection calculations were performed, as described in Section 4.6.7.2 of the SAR, on the assumption that the natural convection valves were open. A power level of 100 kW and a coolant height of 10 feet above the top of the fuel plate (4 inches below overflow) were assumed. The maximum fuel clad temperature was calculated to be below incipient boiling if the pool temperature is maintained below the normal outlet temperature scram point of 60 ˚C.

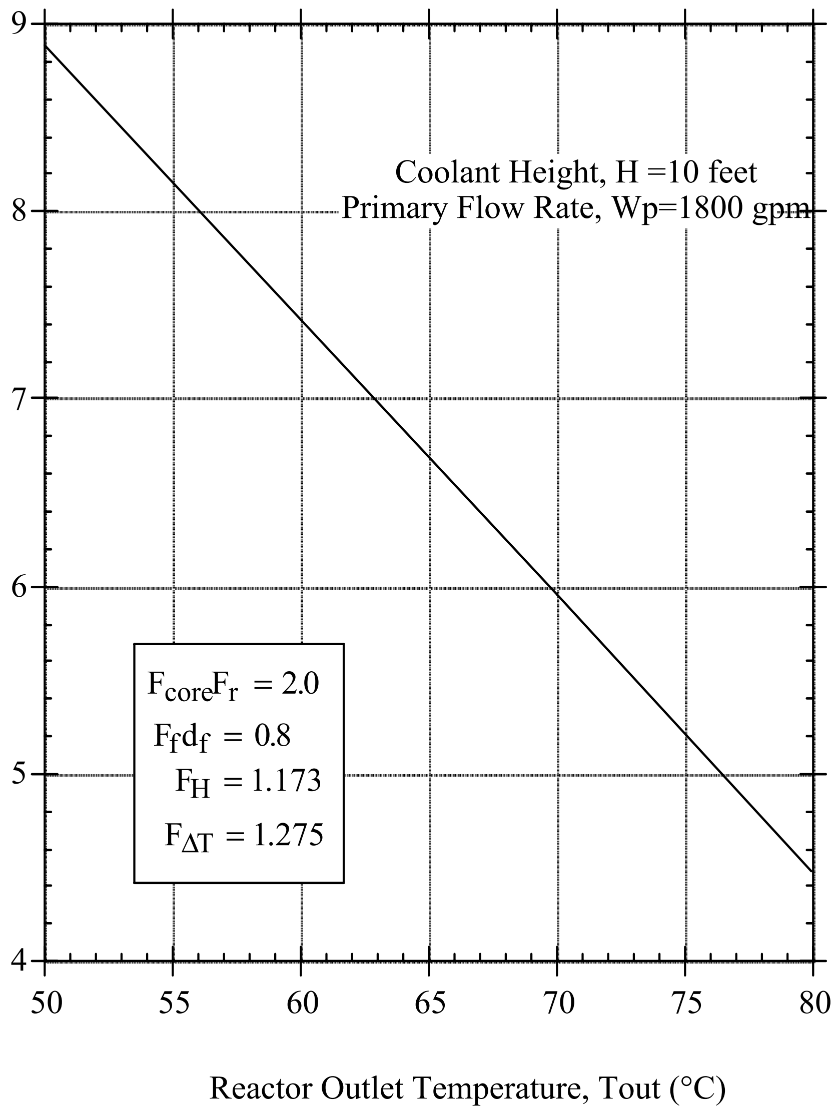

Figure 2.2-1 MITR-II Limiting Safety System Settings for Forced Convection Operation (Two Pumps).

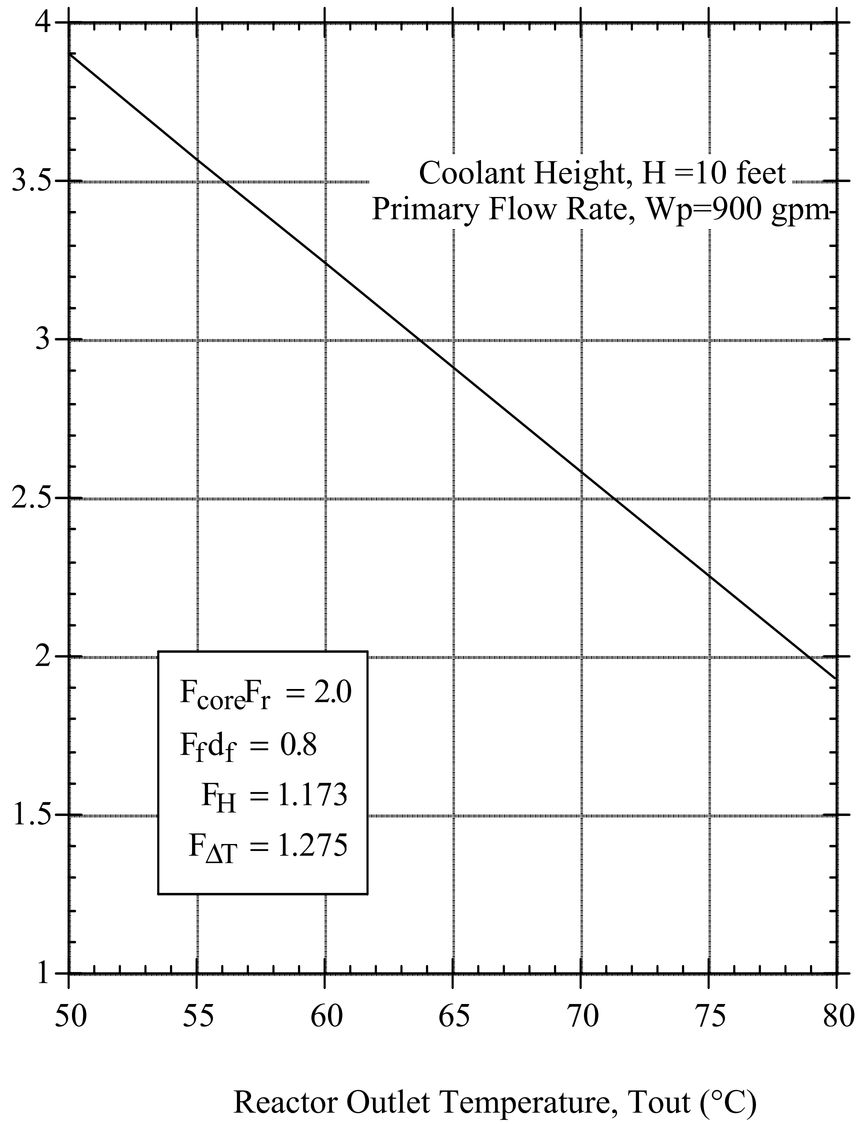

Figure 2.2-2 MITR-II Limiting Safety System Settings for Forced Flow Operation (One Pump).

3. LIMITING CONDITIONS FOR OPERATION

This section of the MITR-II Technical Specifications contains limiting conditions for operation (LCOs). These LCOs are derived from the safety analyses in the SAR, which provide the bases for the LCOs. LCOs are implemented administratively or by control and monitoring circuitry to ensure that the reactor is not damaged, that the reactor is capable of performing its intended function, and that no one suffers undue radiological exposures because of reactor operation.

The LCOs conform to the intent of ANSI/ANS-15.1-2007 as amplified by NUREG-1537, Part I, Rev. 0, 2/96.

3.1 Reactor Core Parameters

3.1.1 Excess Reactivity

Applicability

This specification applies to the subcritical interlock and thereby to the allowed excess reactivity.

Objective

To limit the excess reactivity.

Specification

The reactor shall not be made critical unless the infinite-period, critical shim bank height under cold (10 ˚C), xenon-free conditions shall be greater than 5.0 inches with the regulating rod also at or above 5.0 inches and all movable and non-secured experiments in their most reactive state.

Basis

The basis for the specification is contained in Sections 4.5.3.2 and 4.5.3.3 of the SAR. Observance of the subcritical limit interlock restricts the allowed critical shim bank height to 5.0 inches or greater. This places an upper limit on the excess reactivity that could be inserted upon withdrawal of the blades past the critical position. This use of a physical interlock is preferable to reliance on an administrative limit. As discussed in Section 4.5.3.2 of the SAR, observance of this requirement limits the total excess reactivity in the core to about 9.4 beta. This equates to about 1.6 beta per blade which is the maximum that could realistically be inserted because only one blade can be physically withdrawn at a time.

3.1.2 Shutdown Margin

Applicability

This specification applies to the shutdown margin requirement.

Objective

To ensure that the reactor can be safely shut down at any time.

Specification

The reactor shall not be made critical unless the reactor can be made subcritical using shim blades by at least 1% ΔK/K from the cold (10 ˚C), xenon-free critical condition with the most reactive operable blade and the regulating rod fully withdrawn and with all movable and non-secured experiments in their most reactive state.

Basis

The shutdown margin requirement incorporates the following general philosophy concerning reactor safety:

> It should be impossible for a reactor to be made critical in its most reactive situation on the withdrawal of a single rod. Conversely, it should always be possible to shut down the reactor with one rod stuck in its outermost position. If it is possible that rods or mechanisms might interact so that several could be stuck in the out position, then the number of rods included in the stuck rod criterion should be increased accordingly [3.1.2-1].

The value selected for the shutdown margin (1% ΔK/K) is substantial, and it can be readily determined.

References

3.1.2-1 Thompson, T.J. and J. G. Beckerley (Eds.), The Technology of Nuclear Reactor Safety, Vol. I, the MIT Press, Cambridge, MA (1964), p. 677.

3.1.3 Maximum Safe Step Reactivity Addition

Applicability

This specification limits the amount of reactivity which may be added to the reactor in a single amount.

Objective

To ensure that the integrity of the fuel is maintained during any credible step reactivity excursion.

Specification

1. The reactor shall not be made critical unless the maximum amount of reactivity that may be added to the critical reactor by the credible failure or malfunction of any experiment or component or any set of circumstances which could credibly couple two or more components or experiments shall be less than 1.8% $\Delta K/K$.

Basis

The most limiting case for a credible step (fast ramp) reactivity insertion transient was found to be one initiated at full reactor power (6 MW). It was calculated that the maximum fuel clad temperatures are 83.4 ^oC with a 2.3 beta (1.8% $\Delta K/K$) reactivity insertion over a period of 0.5 s. The cases of low power and with natural convection were also analyzed, but found to be less limiting in that a reactor scram occurred before significant heat could be added.

The step reactivity insertion limits for the MITR-II are chosen to be 2.3 beta. This is conservative given the above analyses. This limit is also conservative relative to ones obtained from correlations with SPERT data [3.1.3-1].

<u>References</u>

3.1.3-1 Safety Analysis Report for the MIT Research Reactor (MITR-II), MITNE-115, October 1970.

3.1.4 Core Configurations

Applicability

This specification applies to conditions pertinent to reactor core design and operation.

Objective

To ensure that reactor core configurations are maintained within the envelope of conditions that were used to establish the thermal hydraulic limits.

Specification

1. The reactor shall not be made critical unless all fuel elements and other core components are secured in position and the hold-down plate is latched in position.

2. The reactor shall not be made critical unless at least five shim blades are operable and any inoperable blade is at the average shim bank height or higher.

3. Pu-Be neutron sources shall not be used in the reactor above a reactor power of 500 watts.

4. The reactor shall not be operated above a power level of 1.0 kW unless:

   a) The safety limit factor,

   $$F_{core}\left(\frac{F_r F_H}{RF_f d_f} - 1\right),$$

   is predicted to be less than 2.4. Definitions of the parameters that comprise the safety limit factor are given in the basis of Specification 2.1.

   b) The core is predicted to operate below incipient boiling at every point in the core by the use of Equation (2.2-1).

   c) Before each refueling or change in core loading which might increase the parameters listed in 4(a) and 4(b) above over a previous operating condition, an evaluation shall be made to ensure that items 4(a) and

4(b) above are satisfied. A record of these evaluations shall be completed and approved by the Reactor Engineer and a second individual who holds a senior operator's license for the MIT Research Reactor and has a minimum of five years of responsible reactor experience. The Reactor Engineer shall have, as a minimum, a BS or higher degree, or equivalent academic experience, in a field of science or engineering that includes the study of reactor neutronics.

d) All positions in the core are filled with either a fuel element or another approved unit. The value of F_f used in the calculation required by 4(a) and 4(b) above shall correspond to the number of non-fueled positions in the core.

5. The reactor shall not be operated at power levels of greater than 100 kW unless:

   a) Primary coolant flow is established.

   b) At least five operable shim blades are within 2.0 inches of the average shim bank height and any inoperable blade is at the average shim bank height or above, except that greater imbalance may exist when one or more shim blades is being inserted to make the reactor subcritical.

   c) The reactor top shield lid is in position.

6. Fuel elements shall neither be inserted nor removed from the core unless the reactor is in a shutdown condition.

7. A shim blade shall not be removed from the core unless the heavy-water reflector is dumped or the shutdown margin criteria will be met with the most reactive blade and the regulating rod fully withdrawn after the control blade has been removed.

Basis

1. Hold-Down Grid Plate

   All fuel elements and core components must be secured in position to prevent mechanical damage of the components, to preclude reactivity changes that might result from inadvertent movement, and to ensure proper flow distribution and cooling.

2. Operable Blades

Section 13.2.9.1 of the SAR discusses operation with a blade below the average bank position. Such operation will not cause a significant change in the core power distribution. Nevertheless, it should be avoided.

3. Pu-Be Neutron Sources

The use of the two one-curie Pu-Be neutron sources has been analyzed and determined to be safe up to a reactor power of 500 watts. This analysis was submitted to the NRC by letter dated March 28, 1975.

4. Safety Calculations

a) Safety Limit Factor

The safety limits, as shown in Figure 2.1-1, are calculated using a value of 2.4 for the safety limit factor. The safety limit factor contains measured thermal hydraulic parameters that are not amenable to continuous monitoring. The safety limits are calculated assuming R=0.86, F_r=2.0, $F_f d_f$=0.8, F_H=1.173, and F_{core}=1.0. Hence,

$$F_{core}\left(\frac{F_r F_H}{R F_f d_f} - 1\right) = 2.4$$

It is shown in Section 4.6.2.2 of the SAR that a value of 0.86 for R, which is the channel outlet subcooling ratio, is a conservative assumption. The value of 2.4 is selected because it is considerably higher than the actual expected value for the operating core and therefore the resulting safety limits are sufficiently distant from the operating range. Specifically, for a safety limit factor of 2.4, the calculated safety limit on the power level is 9.1 MW with the flow and outlet temperatures at their limiting safety system settings (LSSS). An adequate margin is thereby provided between the safety limits and the limiting safety system settings.

b) Incipient Boiling Limit

The overall operating limits to prevent incipient boiling are interrelated by Equation 2.2-1. Evaluation of core operating conditions to prevent incipient boiling should be performed and approved before each refueling operation.

c) Evaluation of the Core Factors and Axial Power Distribution

(i) The values of the flow disparity factor, d_f, were determined for all fuel element positions by a combination of fuel channel tolerance and relative flow measurements during the preoperational testing of the MITR-II. These values are subsequently used for evaluation of the safety limit factor and the incipient boiling limit.

(ii) The nuclear hot channel factor, F_r , which is used to establish the safety limits and the limiting safety system settings, is determined either by experiment or by calculation. In all cases, the nuclear hot channel factor shall be used with a conservative estimate of its uncertainty as described in Section 4.6.4 of the SAR. By definition F_r is the ratio of the power deposited in the hottest channel to that in the average channel. The relative plate power can be derived by experiment, such as gamma scanning of representative fuel plates.

(iii) When experiments or experimental facilities are placed in the core and use part of the primary coolant, the flow rate through them will be determined either by measurement or by calculation such that a conservative value of F_f is obtained.

(iv) The axial power distribution shall be evaluated either by measurement or by calculation. The evaluation of the power distribution involves the determination of the hot spot, which has the highest clad temperature, in the core. The hot spot may or may not be at the point of highest power density, which is usually at the core bottom or coolant inlet end of the channel, because a smaller power density peak that occurs further up the channel where the bulk coolant temperature is higher could lead to the highest clad temperature. In all calculations made for the SAR, it is conservatively assumed that the hot channel has the highest power (maximum F_r) and the lowest flow rate (minimum d_f). Therefore, the limiting safety system settings derived in Section 4.6.7 of the SAR should provide a lower bound of the safe operating conditions. However, the incipient boiling limit will be calculated using Equation (2.2-1) before refueling because the power distributions (axial and radial) are affected by factors such as core configuration, fuel burnup, and shim bank height.

(v) The Reactor Engineer qualifications are made so that appropriate experience is utilized in the evaluation of operating parameters. The "equivalent academic experience" may include nuclear-qualified naval officers and/or MIT graduate students who are pursuing an advanced degree in nuclear fission engineering.

d) The safety limits in Specification 2.1 are derived on the basis of no excessive bypass flow among the fuel elements. All positions in the core must be filled in order to ensure such a flow. The value of F_f depends on the number of non-fueled positions.

5. Reactor Operation at 100 kW or Higher

a) For operation at 100 kW or higher, forced convection is required.

b) The established safety limits assume a banked shim blade height. An imbalance of ± 2.0 inches will not appreciably affect the core power distribution.

c) The reactor top shield is part of the biological shield. To facilitate the performance of various experiments placed in the core, the reactor may be operated at power levels below 100 kW with this shield removed. The total dose rate at 100 kW on the surface of the coolant is approximately 1.3 rem/h. This dose rate is not in excess of those occasionally encountered during certain maintenance operations, and it has been demonstrated that administrative controls can provide adequate control under such conditions. Adequate controls will be instituted during such experiments to prevent excessive personnel exposure.

6. Insertion/Removal of Fuel

Insertion or removal of fuel is done while shut down to ensure safety.

7. Shim Blade Removal

The shutdown margin is modified so that it is met with the two most reactive blades in their full-out positions. This ensures safety. Alternatively, the D_2O reflector may be dumped.

3.1.5 Reactivity Coefficients

Applicability

This specification applies to core reactivity coefficients.

Objective

To ensure that reactivity coefficients are negative.

Specification

Reactivity coefficients (fuel temperature, primary and D_2O temperature, and void) shall be negative over the normal operating range (10 ˚C - 60 ˚C). Any significant observed change in the magnitude of the coefficients in excess of ±20% from the values measured during the MITR-II startup testing shall be evaluated.

Basis

MITR-II reactivity coefficients are discussed in Section 4.5.2.2 of the SAR. All magnitudes are negative for operation above 10 ˚C. (Note: The coolant temperature coefficient is slightly positive below 10 ˚C with the total effect being a few millibeta of reactivity.) Measurements of the coefficients over the life of the MITR-II showed no significant change in magnitude from the values measured during the startup testing.

3.1.6 Fuel Parameters

Applicability

This specification applies to fuel parameters.

Objective

To ensure that the fuel cladding is periodically inspected.

Specification

1. The reactor shall not be operated with damaged fuel except as may be necessary to identify the location of the damaged fuel.

2. Visual inspections of the fuel in the core shall be performed to detect possible deterioration of the clad. The following shall satisfy this requirement:

   a) A visual examination of the core using normal lighting,

   b) A visual examination of the core with normal lighting secured, and

   c) Visual examination of the visible surfaces of each element whenever the element is moved during a refueling.

3. Peak fuel burnup shall not exceed 1.8 x 10^{21} fissions/cm^3 for fuel that is intermetallic UAl_x with 4 to 11% voids.

Basis

1. Damaged Fuel

   The MITR-II uses the fuel sipping method to identify incipient fuel element clad failures [3.1.6-1]. This method requires operation of the core at power for sufficient time to generate detectable activity.

2. Fuel Inspections

The program described in Specification 3.1.6.2 above has been in place for more than a decade and is effective as noted in Section 4.2.1 of the SAR. The visual inspection with normal lighting secured makes use of Cherenkov radiation to backlight the fuel. This makes visible any defect, such as a small blister, that protrudes into a coolant channel. All fuel inspections are done in the core tank under water.

3. Fission Density

The peak fission density limit is based on information developed and tested as part of the fuel designs for the Engineering Test Reactor (ETR) and the Advanced Test Reactor (ATR) [3.1.6-2, 3.1.6-3]. These fuel plate designs and tests cover the range of fuel loading expected for the MITR-II core.

References

3.1.6-1 Clark, L. Jr., Bernard, J.A., and E. Karaian, "Fuel Cladding Failure at the MIT Research Reactor," *Transactions of the American Nuclear Society*, Vol. 38, Suppl. 1, Aug. 1981, pp 25-26.

3.1.6-2 J.M. Beeston, R.R. Hobbins, G.W. Gibson, and W.C. Francis, "Development and Irradiation Performance of Uranium Aluminide Fuels in Test Reactors," *Nuclear Technology* 49, pp. 136-149, June 1980.

3.1.6-3 G.W. Gibson, *The Development of Powdered Uranium-Aluminide Compounds for Use as Nuclear Reactor Fuels*, IN-1133, Idaho Nuclear Corporation, Idaho Falls, Idaho, December 1967.

3.2 Reactor Control and Safety System

3.2.1 Operable Control Devices

Applicability

This specification applies to the reactor control and protection systems.

Objective

To specify the number and type of operable control and safety devices as well as the allowed scram times.

Specification

1. The reactor shall not be made critical unless:

   a) There are six shim blades, each with a scram capability, and one regulating rod. A minimum of five shim blades shall be operable as stipulated in Specification 3.1.4.2.

   b) The time from initiation of a scram signal and movement of each operable blade from its current position to its 80% inserted position is less than one second for each blade.

Basis

The MITR-II is equipped with six shim blades and one regulating rod. The blades are connected to their drives by electromagnets and hence can be dropped into the core upon initiation of a scram signal. The basis of the second part of the specification is industry practice. A scram time of 1.0 second is the standard. Analyses in the SAR show that a 1.0 second scram time will not result in any damage to the fuel. (Note: For some analyses, a 2.0 second delay was assumed as a conservative measure.)

3.2.2 Reactivity Insertion Rates and Automatic Control

Applicability

This specification applies to the reactivity control system.

Objective

To ensure that the integrity of the fuel is maintained during any credible ramp reactivity excursion.

Specification

1. The maximum controlled reactivity addition rate is no more than 5×10^{-4} $\Delta K/K/s$.

2. Only one shim blade shall be withdrawn at a time.

3. Shim blades and/or the regulating rod may be connected to automatic controllers within the limitation of Specification 3.2.2.4 below.

4. The total available positive reactivity of any control device connected to an automatic controller, other than those covered by Specification 6.4, shall be less than 0.5% $\Delta K/K$.

5. The nuclear safety system shall be separate from any automatic controller.

Definitions

The following facility-specific definitions are provided relative to the above specification:

1. Total Available Positive Reactivity

   The "total available positive reactivity" of any control device connected to an automatic controller is the positive reactivity beyond the critical condition that could be inserted if the control device were fully withdrawn.

2. Separate

   The word "separate" means that the output of an instrument used in the safety system will not be influenced by interaction with the control system. For example, a signal derived from an instrument that forms part of the safety system would not be transmitted to the control system unless first passed through an isolation device.

Basis

The basis of Specification 3.2.2.1 is discussed in Section 13.2.2.2 of the SAR where ramp reactivity insertions are analyzed. The control blade and regulating rod speeds are designed to limit the reactivity addition rate to less than 5×10^{-4} ΔK/K/s. This value is conservative within the range of reactivity insertion rates normally accepted for reactor operation. Control systems in this range give ample margin for proper human response during approach to criticality and power operation. In the event of an accidental continuous insertion of reactivity at this maximum rate, the response of the reactor safety system period and level trips will adequately protect the reactor. The MITR-II's control system is constructed so that only one shim blade can be withdrawn at a time.

The basis of Specification 3.2.2.2 is that the simultaneous withdrawal of two or more blades is not physically possible.

The basis of Specifications 3.2.2.3 and 3.2.2.4 is discussed in Section 10.3.2.8 of the SAR. There are two methods for ensuring the safety of automatic controllers, either analog or digital. One is to limit the reactivity worth of the control device. The other is to design the controller to incorporate the property of "feasibility of control." The first of these options is addressed here. The second is addressed in Specification 6.4. Controllers designed under this Specification 3.2.2 are governed by (1) a limitation of the rate of reactivity insertion, (2) a limitation on the reactivity worth of the associated absorber, (3) a requirement that the capability of the safety system to perform its function is not impaired. The value chosen for the reactivity worth limitation is 0.5% $\Delta K/K$ which is well below the step reactivity insertion limit.

3.2.3 Reactor Protection System

Applicability

This specification applies to the reactor protection system.

Objective

To ensure that automatic protection action is provided as required by the reactor protection system.

Specifications

1. The reactor shall not be made critical unless the reactor protection system is operable in accordance with Table 3.2.3-1.

2. Fuel shall not be moved and no work involving reactivity shall be performed in the core unless the period and neutron flux level channels are set to alarm within the zero primary pump limits of Table 3.2.3-1. In addition, the manual major scram is operable for building isolation and the D_2O dump valve selector switch is operable unless the D_2O reflector is already dumped.

3. The reactor shall not be made critical unless scram setpoints are set more conservatively than the corresponding LSSS.

Table 3.2.3-1

Required Safety Channels

	Channel / Parameter	Action	2 Primary Pumps Limiting Setpoint	2 Primary Pumps Min. No. Required	1 Primary Pump Limiting Setpoint	1 Primary Pump Min. No. Required	0 Primary Pump Limiting Setpoint	0 Primary Pump Min. No. Required
1.	Period	Scram	> 7 sec.	3[(1)(5)]	> 7 sec	3[(1)(5)]	> 7 sec	3[(1)(5)]
2.	Neutron flux level	Scram	< 7.4 MW	3[(1)(5)]	< 3.2 MW	3[(1)(5)]	< 100 kW	3[(1)(5)]
3.	Low count rate	Scram	> 5 cps	3[(1)(5)]	> 5 cps	3[(1)(5)]	> 5 cps	3[(1)(5)]
4.	Primary coolant outlet temperature	Scram	< 60° C	2	< 60° C	2	< 60° C	2
5.	Core tank level	Scram	4" below overflow pipe	1	4" below overflow pipe	1	4" below overflow pipe	1
6.	Reflector tank level	Scram	4" below overflow	1	4" below overflow	1	4" below overflow	1[(2)]
7.	D_2O dump valve selector switch	Reflector dump & scram	N/A	1	N/A	1	N/A	1
8.	Manual major scram	Reflector dump, containment closure & scram	N/A	2[(3)]	N/A	2[(3)]	N/A	2[(3)]
9.	Manual minor scram	Scram	N/A	1	N/A	1	N/A	1
10.	Primary coolant flow rate	Scram	> 1800 gpm	2[(4)]	> 900 gpm	2[(4)]	N/A	0
11.	D_2O reflector flow rate	Scram	> 75 gpm	1	> 75 gpm	1	N/A	0
12.	Shield coolant flow rate	Scram	> 50 gpm	1	> 50 gpm	1	N/A	0

1) Nuclear safety scram logic system ensures that reactor scrams when two trips are present simultaneously from any two of the four nuclear safety channels.
2) For reflector reactivity measurement, the reflector scram can be bypassed at power levels less than 100 kW.
3) One in utility room.
4) At least one safety channel on the primary coolant flow rate scram must be by core inlet pressure sensor.
5) Within 15 minutes of declaring any nuclear safety system channel inoperable, the channel must be placed into a tripped state, which will be indicated on the Safety System Condition LED Scram Display. If any nuclear safety channel is in a tripped state, and a second nuclear safety channel is declared inoperable, then within 15 minutes at least one of the two must be returned to an operable state or the reactor must be shut down.

Table 3.2.3-1 (Continued)

Required Safety Channels

	Any Number of Pumps (Two, One, or Zero)			
	Channel / Parameter	Action	Setpoint	Minimum No. Required
13.	Nuclear safety channel in test or fault	Scram	Channel in test or fault condition	3[(1)(5)]
14.	Building overpressure	Scram	< 3" water above atmospheric	1
15.	Main personnel lock gaskets deflated	Scram	Both gaskets deflated	1
16.	Basement personnel lock gaskets deflated	Scram	Both gaskets deflated	1
17.	Hold-down grid unlatched	Scram	Grid unlatched	1
18.	Experiment scrams	(As Required by Experiment Approval)		

1) Nuclear safety scram logic system ensures that reactor scrams when two trips are present simultaneously from any two of the four nuclear safety channels.
5) Within 15 minutes of declaring any nuclear safety system channel inoperable, the channel must be placed into a tripped state, which will be indicated on the Safety System Condition LED Scram Display. If any nuclear safety channel is in a tripped state, and a second nuclear safety channel is declared inoperable, then within 15 minutes at least one of the two must be returned to an operable state or the reactor must be shut down.

Basis

The nuclear safety system, consisting of four wide-range nuclear safety channels, provides protection against high power level and short reactor period. Each nuclear safety channel produces a trip signal on high power, short period, low detector count rate, channel in test, or channel fault / equipment malfunction. The scram logic system downstream will scram the reactor upon any simultaneous combination of these trips from two of the four nuclear safety channels, thereby ensuring there are three operable channels whenever the reactor is not shut down. The system is designed to allow the reactor to be critical with one channel in an inoperable state, including removal of the channel for service or maintenance; should any one of the remaining three required channels produce a trip, the scram logic system will scram the reactor. Any nuclear safety channel declared inoperable will be placed into a tripped state as indicated on the LED Scram Display. The 15 minute time allowance is based on the time needed to make the necessary notifications.

The nuclear safety system is required at all power levels including certain subcritical operations such as refueling, absorber change-out, or other in-core work that affects reactivity. Above 100 kW, protection is also required on primary, D2O, and shield coolant flows.

The parameters listed in Table 3.2.3-1 are monitored by the reactor protection system. This system automatically initiates action to ensure that appropriate limiting safety system settings and limiting conditions of operation are not violated.

In practice, low power physics tests including rod reactivity worth measurements are usually performed at power levels of less than 10 kW and in the absence of forced convection primary flow. The upper limit of 100 kW for this type of operation was established on the basis of adequate natural convection cooling. The maximum plate temperature at 100 kW with natural convection cooling is estimated to be below incipient boiling, if the coolant outlet temperature is maintained below the normal scram point of 60˚C. Therefore, the reactor outlet temperature channel is specified in Table 3.2.3-1 as 60˚C for zero pump operation.

The reflector tank low D_2O level scram must be bypassed during low power operation if calibration of the reactivity effect of the D_2O reflector dump safety system is to be performed.

For refuelings, the reactor is in a shutdown condition, primary flow is secured, and the D_2O reflector is normally dumped. Therefore, the nuclear safety channels are set to alarm within the zero primary pump limits for period and level. The capability to isolate the building is required. This is provided by the major scram. Finally, it should be possible to dump the D_2O reflector, if it is not already dumped.

3.2.4 Control System Interlocks

Applicability

This specification applies to the reactor control system.

Objective

To ensure that interlocks that prevent control device withdrawal are incorporated in the control system as appropriate.

Specification

1. The following interlocks shall be operable, except as noted in Specification 3.2.4.2 and 3.2.4.3 below, before the reactor is made critical:

   a) Withdraw Permit (Startup) Interlock - Shim blade electromagnets cannot be energized unless all required safety channels in the reactor protection system and all startup interlocks are satisfied.

   b) Subcritical Limit - Shim Blades Interlock - Shim blade cannot be withdrawn above 5.0 ± 0.5 inches unless each blade is first raised to that height.

2. The "Subcritical Limit - Shim Blade Interlock" may be bypassed during critical operation (power level ≤ 100 kW) for the purpose of measuring the reactivity worths of the shim blades and the regulating rod.

3. The "No Overflow Reflector Startup" interlock and the "Low Level D_2O Reflector" scram may be bypassed at power levels of less than 100 kW for the purpose of reflector reactivity measurement.

Basis

The basis for the specification is given in Section 7.3.1 of the SAR. Closure of the withdraw permit circuit means that both the reactor protection system scrams and the startup interlocks (all absorbers full in, no overflow core tank and no overflow reflector tank clear, and building ΔP) are satisfied.

Satisfaction of the subcritical interlock assists the operator in establishing a uniform bank height prior to the final approach to criticality. Reactivity worth measurements require that this interlock be bypassed because the blades cannot be kept at an even bank height. Also the "No Overflow Reflector Startup" interlock and "Low Level D_2O Reflector" scram must be bypassed during reflector reactivity measurements because it is necessary to vary the level of the reflector.

3.2.5 Backup Shutdown Mechanisms

Applicability

This specification applies to the heavy-water reflector system.

Objective

To ensure a backup means of shutdown for the reactor.

Specification

1. The reactor shall not be made critical unless the reactivity worth of the D_2O reflector dump is greater than the reactivity worth of both the most reactive blade and the regulating rod.

2. The level in the D_2O reflector dump tank shall not exceed 25 inches whenever the reactor is critical. This requirement does not apply for reflector reactivity measurement at power levels less than 100 kW.

Basis

The basis is given in Section 4.5.3.1 of the SAR. The capability to dump the D_2O reflector serves as a diverse means of reactor shutdown and hence the reactivity worth of that action should exceed the worth of both the most reactive shim blade and the regulating rod because the definition of shutdown margin assumes that these two devices are stuck in their full-out position on shutdown. No analysis in the SAR relies upon an assumption that dumping of the reflector provides protection against a transient condition. Hence, there is no limit on the time for a D_2O reflector dump. The limit on level of the dump tank ensures that there is sufficient free volume in the tank to accommodate the reflector dump.

3.2.6 Bypassing Channels

Applicability

This specification applies to the reactor safety system.

Objective

To ensure that required safety channels are not bypassed during reactor operation.

Specifications

The safety channels and interlocks listed in Specifications 3.2.3 and 3.2.4 as required for critical operation shall not be bypassed during critical operation of the reactor except as noted in those Specifications.

Basis

Refer to the bases of Specifications of 3.2.3 and 3.2.4.

3.2.7 Control Systems and Instrumentation Requirements for Operation

Applicability

This specification applies to the reactor control system and to the control console display instrumentation.

Objective

To ensure that the console operator has sufficient indication of power level, reactor period, primary coolant flow, primary coolant outlet temperature, core tank level, and control device position.

Specification

Indication from the instrumentation listed in Table 3.2.7-1 shall be provided to the reactor console operator prior to reactor startup and during reactor operation.

Table 3.2.7-1

Required Instrumentation for Display

Parameter	**Minimum Number**	**Location**
1. Period	1	Console
2. Neutron Flux Level		
a) Wide Range	1	Console
b) Linear Power	1	Console
3. Core Tank Level	1	Control Room
4. Primary Coolant Flow	1	Control Room
5. Coolant Outlet Temperature	1	Control Room
6. Shim Blade Position(1)	5	Console
7. Regulating Rod Position(2)	1	Console

(1) Indication required for all operable shim blades. Indication may be either numeric or analog meter or both.

(2) Indication may be either numeric or analog meter or both.

Basis

The basis of this specification is given in Section 7.4 of the SAR. The limiting safety system settings are a function of the reactor power, coolant flow, coolant temperature, and core tank level. These parameters, together with reactor period, are important to safe operation. Indication of shim blade position is also important. There are four independent nuclear safety channels, each of which monitors wide-range reactor power level and period. The operator requires continuous indication of reactor power, reactor period, and control device position in order to perform power manipulations. Hence, these parameters are displayed on the reactor console. The operator requires knowledge of whether or not flow, temperature, and level are within their normal ranges. Hence, they are displayed in the control room but not necessarily on console.

3.3 Coolant Systems

3.3.1 Natural Convection and Anti-Siphon Valves

Applicability

This specification applies to the primary coolant system.

Objective

To ensure that core bypass flow is within design limits and to ensure a safe transition from forced circulation cooling to natural convection cooling for decay heat removal.

Specification

1. The natural and anti-siphon convection valves shall be verified closed prior to operation above 100 kW if the primary coolant pumps have been off. This check may be made visually or by use of stethoscope or other similar device capable of detecting valve closure.

2. The reactor shall not be made critical unless:

   a) The natural convection valves are capable of automatically opening to establish natural convection cooling of the core in the event that forced circulation of the coolant stops.

   b) The anti-siphon valves are capable of opening automatically to disrupt a siphon that might form if the primary coolant inlet pipe were to break.

3. The coolant level in the core tank shall be maintained as stated in Specification 2.2 whenever the reactor is critical. At other times, the coolant

level shall be maintained at or above the level of the anti-siphon values, unless there is no fuel within the core tank.

Basis

Operation of the natural convection and anti-siphon valves is described in Sections 6.2 and 6.3 of the SAR. Closure of these valves ensures that primary coolant flow does not bypass the core in amounts greater than anticipated in the derivation of the limiting safety system settings. Both sets of valves promote the establishment of natural convection cooling upon loss of forced convection. In addition, the anti-siphon valves prevent siphoning of the coolant should there be a break in the primary coolant inlet pipe.

The coolant level in the core tank should be maintained at overflow whenever possible for the reasons stated in Section 5.2.1.1 of the SAR. However, for some maintenance operations, such as blade drive changes, it is necessary to lower the level. This is done in accordance with written, approved procedures that include a requirement to monitor radiation levels.

3.3.2 H_2 Concentration Limit

Applicability

This specification applies to the H_2 gas concentration in the air space above the core.

Objective

To prevent a flammable concentration of H_2 gas.

Specification

1. The H_2 concentration in the air space above the core shall not exceed 3.5 volume percent.

2. In the event of isolation of the air space above the core for more than five minutes, reactor power shall be reduced to < 100 kW.

Basis

The basis of this specification is given in Section 5.2.1.12 of the SAR. The minimum explosive concentration for mixtures of H_2 in the air is given as 4.1% by Ref. 3.3.2-1.* A limit of 3.5% is therefore conservative.

In normal operation, with a continuous purge of the air space, no buildup of radiolytic gases will occur. However, if abnormal radioactivity is detected in the purge stream, the air space above the pool will be isolated automatically. The response to such an occurrence would be for the operator to investigate the cause of the isolation and to open the air space and resume the continuous purge as soon as possible. If this is not possible, the reactor should be reduced to

* "Water vapor (up to saturation) and pressure up to 200 atmospheres do not affect limits. Temperatures up to 400° C do not change lower limit." [3.3.2-2]

< 100 kW within five minutes. Continued reactor operation at <100 kW is considered acceptable because the rate of H_2O decomposition at this low power level is insignificant.

References

3.3.2-1 Lewis, R.J. Jr., "Sax's Dangerous Properties of Industrial Materials," 8th edition, Van Nostrand Reinhold Publishing Company, New York, 1994.

3.3.2-2 Weintraub, A.A., "Control of Liquid Hydrogen Hazards at Experimental Facilities," HASL-160. Health and Safety Laboratory, New York Operations Office, AEC, May 1965.

3.3.3 D_2 Concentration Limit and Recombiner Operation

Applicability

This specification applies to the D_2 gas concentration in the helium blanket above the D_2O in the reflector system and to the operation of the recombiner system in the D_2O reflector system.

Objective

To prevent a flammable concentration of D_2 gas in the helium blanket.

Specification

1. The D_2 concentration in the helium blanket shall not exceed 6 volume percent.

2. Before increasing reactor power above 100 kW and at least every two hours during operation above 100 kW the temperature in the middle of the recombiner shall be verified to be >50 ˚C. In addition, the recombiner flow rate shall be verified >1.5 and <8 cfm prior to reactor startup if the reactor has been shut down for more than 24 hours.

3. If either of the parameters listed in Specification 3.3.3.2 above falls outside the above limits and cannot be corrected within a fifteen minute period, reactor power shall be reduced to < 100 kW.

Basis

The basis of this specification is given in Section 5.3.1.12 of the SAR. Recombination of the disassociated D_2 and O_2 is accomplished by continuously circulating the helium from above the reflector through a catalytic recombiner. The flow through the recombiner is held at approximately two cubic feet per minute, and the recombiner operates at a temperature above 50 ˚C as measured at the middle of the reaction chamber. A rise in temperature in the recombiner is a result of the recombination process and is positive indication that the recombiner is operating properly.

In a report, "Flammability of Deuterium in Oxygen-Helium Mixtures," issued by the Explosives Research Center of the Bureau of Mines [3.3.3-1], it is shown that the volume percent of D_2 needed for flammability is independent of the volume percent of O_2 from 4 to 30%. The data in this report give the flammable concentration of D_2 as 7.8 volume percent at 25 ˚C and 7.5 volume percent at 80 ˚C. Extrapolation of these two points by a straight line approximation indicates a flammable concentration of 6.8 volume percent at a temperature of 200 ˚C. These results are conservative because ignition in the tests was initiated at the base of the combustion tube.

The maximum temperature in the helium system will be less than 200 ˚C under all foreseeable circumstances. Hence, it can be concluded that combustion will not occur if the D_2 concentration is kept less than 6 volume percent.

If the recombiner operation falls outside the above specified range and cannot be corrected within a fifteen minute period, the reactor power shall be reduced to <100 kW. Continued reactor operation at <100 kW with the recombiner out of service is considered acceptable because the rate of D_2O decomposition at this low power level is insignificant.

References

3.3.3-1 "Flammability of Deuterium in Oxygen-Helium Mixtures," TID-208998, Explosives Research Center, Bureau of Mines, June 15, 1964.

3.3.3-2 File Memo (H_2/D_2 Concentration Measurements)

3.3.4 Emergency Cooling Requirements

Applicability

This specification applies to the emergency cooling system including pipes, valves, and spray nozzles.

Objective

To ensure that adequate core cooling is provided to prevent fuel overheating in the event of a complete loss of coolant from the main core tank.

Specification

1. The reactor shall not be operated at power levels in excess of 100 kW unless the emergency core cooling system is operable and capable of providing the reactor core with a minimum total emergency cooling flow rate of 10 gpm within 5 minutes after a low level core tank scram.

2. The reactor shall not be operated at power levels in excess of 100 kW unless the emergency core cooling spray nozzles are positioned so that each fuel element will receive at least 20% of the average flow, which is defined as the total emergency cooling flow divided by the number of fuel elements.

Basis

The basis of this specification is given in Section 6.4 of the SAR. The complete loss-of-coolant accident requires a simultaneous massive rupture of the core tank and the reflector tank. A second type of loss-of-coolant accident would be from the rupture of the inlet pipe below the core level, together with a failure of the anti-siphon valves to open.

The primary safeguard against fuel overheating because of these extreme failures is a provision to spray cool the core with an independent continuous water supply until such time as air-cooling is adequate or until recirculation can be initiated through other means. The emergency spray system is a redundant system. Two independent connections to the city water, including spray nozzles, piping, and valves are available, either of which will satisfy the emergency cooling requirement. This system is described in Section 6.4 of the SAR, where it is shown that 9.5 gpm is adequate flow to remove the heat generated at any time after shutdown. As discussed in the SAR, the nozzles are positioned so that each element receives at least 20% of the average spray flow. The system does not depend on the availability of normal electrical power.

3.3.5 Coolant Radioactivity Limits

Applicability

The specification is applicable to the primary, D_2O, and secondary coolants.

Objective

To ensure detection of deterioration of components in the reactor coolant systems and to identify leakage in heat exchangers.

Specification

1. The primary coolant shall be analyzed for gross activity and for isotopic identification. If the primary coolant activity exceeds 3 times the nominal fission product activity, primary coolant sampling frequency shall be increased and action shall be initiated in accordance with Specification 3.1.6 to determine if any in-core fuel element is damaged and with Specification 3.3.6 to determine if water chemistry requirements are being met. Reactor operation may continue if the cause of the problem is known and if such operation is allowed pursuant to Specifications 3.1.6 and 3.3.6.

2. The D_2O reflector coolant shall be analyzed for gross activity and tritium. The radioactivity of the tritium in the D_2O coolant should not exceed 5 mCi/ml. If the radioactivity of the tritium in the D_2O coolant approaches this guideline limit, preparations shall be initiated to replace the D_2O coolant. Reactor operation may continue while these preparations are in progress, provided that tritium concentration does not exceed 6 mCi/ml.

3. The secondary coolant shall be analyzed for gross activity and for tritium. The tritium activity shall be in accordance with Specification 3.7.2.2. If not, action shall be taken as required by the specification. In addition, the presence of other detectable activity in excess of 10 CFR 20 (Appendix B, Table 3, 'Sewer Disposal') limits in the secondary coolant shall require isolation of the affected heat exchanger. Reactor operation may be continued only as necessary to identify the affected heat exchanger.

Basis

The basis of this specification is given in Sections 5.2.1.11, 5.3.1.11, 5.4.1.11 and 5.5.1.5 of the SAR. Core performance is monitored continuously by the core purge detector which is addressed by Specification 3.7.1.3. The primary coolant sample analysis serves as a backup to the indication provided by that monitor. Also, it provides a means for the detection of trends. An elevated activity level could be the result of a damaged fuel element or it could be the result of an activated impurity. The former would require a reactor shutdown. The latter would require verification that the primary cleanup system was functional. Accordingly, if the guideline limit is exceeded, the actions required by Specifications 3.1.6 and 3.3.6 will be initiated. Also, the primary ion column will be evaluated for operability.

The principal concern of the D_2O reflector coolant system is tritium which builds up slowly over many years. The system is closed and hence the tritium that is contained within it does not pose a hazard during normal operation. However, the radiological controls needed to perform maintenance activities become greater as the activity levels increases. Accordingly, a guideline tritium activity limit of 5 mCi/ml is established along with the requirement to initiate preparations for changing the reflector coolant if the tritium activity approaches this guideline limit. Reactor operation may continue while preparations, which can take many months, are progressing.

The secondary coolant is continuously monitored for activity by on-line detectors (Specification 3.7.1.6). These detectors do not sense tritium because it emits a very low energy beta particle. Accordingly, either a daily analysis is done for tritium or an in-line tritium monitor is in service if secondary coolant is supplied to a D_2O heat exchanger (main or cleanup). If tritium levels exceed that permitted by Specification 3.7.2.2, corrective action is to be in accordance with that specification. In addition, if other detectable activity in excess of 10 CFR 20 limits is identified, the affected heat exchanger will be identified and removed from service.

3.3.6 Primary Coolant Quality Requirements

Applicability

This specification applies to the pH and conductivity, and activity of the primary coolant.

Objective

To control corrosion of the fuel, core components, and primary coolant loop structure as well as to minimize activation of impurities and to maintain visual clarity of the coolant.

Specification

1. The pH of the primary coolant shall be kept between 5.5 and 7.5.

2. The conductivity of the primary coolant, averaged over 24 hours, shall be kept less than 10 μS/cm at 20˚C.

3. Operation of the reactor with the pH or conductivity outside the limits given in Specifications 3.3.6.1 and 3.3.6.2 above is permitted provided:

   a) The pH is between 5.0 and 8.0,

   b) The increase in conductivity is not the result of a chloride ion concentration in excess of 6 ppm,

   c) Sampling of the coolant for pH, conductivity, and chloride ion concentration is done at least once every eight hours, and

   d) The pH band specified in Specification 3.3.6.1 is re-established within 48 hours.

4. The pH of the fuel storage pool shall be kept between 5.5 and 7.5. The conductivity of the fuel storage pool, averaged over 24 hours, shall be less

than 10 μS/cm at 20˚C. If these limits are not met, then sampling frequency shall be increased to at least weekly.

Basis

The basis of this specification is given in Section 5.2.2.1 of the SAR. The purpose of pH monitoring is to ensure that corrosion of the fuel, core components, and the primary coolant loop structure is maintained within an acceptable limit. The fuel cladding and the core tank are made of aluminum alloy. A portion of the primary coolant loop is constructed of stainless steel. Lower pH will reduce aluminum alloy corrosion and oxide film formation on the fuel surface and higher pH is favored to control stainless steel corrosion. Thus, a pH range between 5.5 and 7.5 is selected for the primary coolant.

Electrical conductivity is also monitored to control purity of the primary coolant. A limit of 10 μS/cm is adopted because this limit has been observed for the MITR since 1974 and no corrosion issues have ever been identified with either the fuel or the core structural materials.

The limits on the fuel storage pool water are those that have been observed for the MITR since 1974 and no corrosion issues have ever been identified with either the spent fuel or the storage pool tank.

Operation with out-of-specification chemistry is acceptable for short intervals. The important factors are pH and, if a heat flux is present, the absence of a high chloride ion concentration. A high conductivity by itself is not of concern.

3.4 Reactor Containment Integrity and Pressure Relief System

Applicability

This specification applies to the reactor containment building and to its pressure relief systems.

Objective

To minimize the consequence of a possible release of airborne radioactive effluent from the containment building and to protect its integrity.

Specification

1. Containment integrity shall be maintained, except as noted in Specification 3.4.2 below, when any of the following conditions exist:
    a) The reactor is not secured, or
    b) Movement of irradiated fuel is being performed, except when the fuel is in a properly sealed and approved shipping container, or
    c) Work involving radioactive samples is in progress in one of the reactor floor hot cells.

2. The requirement for containment integrity may be omitted during the performance of tests to ensure the operability of the airlock gasket deflated scrams. These tests shall be done with the reactor in a shutdown condition and Specifications 3.4.1(b) and 3.4.1(c) above shall be observed.

3. Whenever containment integrity is required, the leakage rate of the building containment shall be less than 1% of the containment volume per day per psig overpressure when the ventilation dampers are closed and the pressure relief system valves are sealed.

4. Whenever containment integrity is required, the efficiency of the charcoal absorber in the pressure relief system shall be at least 95% for the removal of iodine.

5. There shall be a vacuum relief system to relieve pressure when the atmospheric pressure exceeds reactor building pressure by greater than 0.1 psig.

6. The following is the minimum equipment required to establish containment integrity.

   a) The main and basement personnel locks are either operable or at least one door is closed with gaskets inflated.

   b) The truck airlock inner door is closed.

   c) All containment penetrations (ventilation, pneumatic, gas supply, electrical) are either sealed or equipped with an isolation device. (Note: If an isolation device is redundant, then only one set of the redundant devices is required.)

   d) All piping penetrations within the reactor building are capable of withstanding containment building test pressure.

   e) Initiation system for containment isolation is operable.

   f) At least one set of the redundant vacuum relief breakers is operable.

   g) Pressure relief system is operable.

Basis

The basis of this specification is given in Section 6.5.1 of the SAR. The reactor containment is an engineered safety feature that serves as the final physical barrier to the release of radioactive particulates and gases. Proper operation of the containment, therefore, is required during all operations that could result in radioactive releases. These include reactor operation, fuel movements, and work with radioactive samples in the reactor floor hot cells.

In order to test the airlock gasket deflated scrams, it is necessary for the reactor to be in a shutdown instead of a secured condition. This activity requires about fifteen minutes per test.

The specified containment leak rate has been used in Chapter 13 of the SAR to evaluate the consequences of the maximum hypothetical accident (MHA).

The building pressure relief system provides a method for controlled depressurization of the reactor containment after initial assessment of any contained radioactivity. The specified efficiency of 95% for the charcoal filter together with the specified containment leak rate were used in the MHA analysis to show that the use of the pressure relief system under accident conditions will not lead to significant increased doses at the site boundary.

The vacuum relief system protects the containment building against underpressure.

3.5 Ventilation System

Applicability

This specification applies to the containment building ventilation system.

Objective

To ensure that the release of airborne radioactivity is within 10 CFR (Section 20.1301 – Dose Limits for Individual Members of the Public) requirements.

Specification

1. Whenever the reactor is operating at power levels in excess of 250 kW, the ventilation system shall be operating to provide at least 7500 cfm of exhaust ventilation flow through the containment building stack.

2. If the reactor is operating at power levels in excess of 250 kW and ventilation is lost, operation may continue for a maximum of five minutes. Reactor power shall then be reduced to less than 250 kW unless ventilation has been restored.

3. An interlock shall be operable that allows reactor startup only when the containment pressure is below atmospheric pressure by at least 0.1 inch of water.

4. The following is the minimum equipment required for operability of the ventilation system:

    a) Intake and exhaust fans.

    b) Auxiliary fans if needed for radiological work.

c) At least one set of the redundant vacuum relief breakers is operable.

d) Controls (manual or remote actuator plus damper) to adjust building differential pressure.

e) Exhaust filters.

f) Ventilation system interlocks listed in Specification 4.5.

g) One gaseous and one particulate radiation monitor located in the ventilation effluent.

<u>Basis</u>

Section 9.1 of the SAR describes the containment building ventilation system. A minimum exhaust flow rate is specified to preclude the buildup of radioactive gases, primarily Ar-41. The ventilation system is interlocked so that if the building differential pressure (relative to the atmosphere) becomes excessive, a trip will occur. Such trips may occur as the result of weather fronts. Accordingly, operation without ventilation is permitted for short intervals in order to allow for the adjustment of the stack exhaust damper and the restoration of ventilation.

Argon-41 production is minimal below 250 kW. Hence, reactor power shall not be raised above 250 kW unless ventilation has been established and reactor power shall be reduced to less than 250 kW within five minutes if ventilation is lost.

It is intended to operate the reactor with a negative containment ΔP. An interlock must be satisfied requiring a minimum 0.1 inch water pressure differential (negative) in the building before a reactor startup can be conducted. This condition ensures that the building containment ΔP is negative at the time of startup. This interlock forms part of the withdraw permit circuit.

3.6 Emergency Power

Applicability

This specification applies to the emergency electrical power system.

Objective

To ensure that loads important to safety are supplied by emergency electrical power upon loss of normal offsite electrical power.

Specification

1. Emergency electrical power with the capacity to operate the equipment listed in Table 3.6-1 shall be available when the reactor is operating and shall be capable of operation for at least one hour following a loss of normal electrical power to the facility.

2. Startup of the emergency power system and the transfer of all loads, except the primary coolant auxiliary pump, to the emergency power system shall be automatic.

3. The following is the minimum equipment needed for operability of the emergency power system:

   a) Batteries sufficient to fulfill the requirement of paragraph (1) above.

   b) A motor-generator set.

   c) Startup circuitry and automatic transfer switches to fulfill the requirement of paragraph (2) above.

   d) A manual transfer switch for the primary coolant auxiliary pump.

Table 3.6-1

Minimum Equipment to be Supplied by Emergency Electrical Power

1. One neutron flux level channel.
2. Core tank coolant level indicator.
3. Primary coolant outlet temperature.
4. Radiation monitors required by Specification 3.7.
5. Containment intercom system.(1)
6. Primary coolant auxiliary pump.
7. Lighting as required for personnel safety.(2)

Note: (1) Alternately, telephones (which are all on MIT's emergency power) may be substituted.

(2) Alternatively, self-contained battery operated lights may be substituted.

Basis

The basis of this specification is given in Section 8.2 of the SAR. The use of emergency power is probably not necessary for the MITR-II because loss of normal electrical power automatically scrams the reactor and because natural convection is sufficient to remove decay heat. Nevertheless, provisions are made for emergency power to supply both a minimum set of instruments and the auxiliary pump. The information supplied to the operator will ensure an orderly procedure in all such cases. The availability of the auxiliary pump will facilitate decay heat removal. The choice of a minimum one hour is based on providing information to the operator for a sufficiently long period following the scram to ensure that the core is receiving adequate cooling.

3.7 Radiation Monitoring Systems, Effluents, Hot Cells, and Byproduct Material

3.7.1 Monitoring Systems

Applicability

This specification applies to the radiation monitoring system.

Objective

To ensure that facility personnel are alerted to the presence of radioactivity, both on site and in effluent paths, and to ensure that the engineered safeguard features that preclude the discharge of radioactivity are operable.

Specifications

1. Whenever the reactor containment building is occupied, there shall be an operable continuous air monitor that has an audible alarm and a means of recording data.

2. Whenever the containment is not isolated and containment integrity is required, the following shall be provided:

   a) The main and exhaust ventilation dampers shall be interlocked with a plenum effluent monitor so that the dampers close upon detection of abnormal levels of radioactivity. This monitor shall indicate and alarm in the control room. The time for this closure to occur shall be less than the transit time of the exhaust air to traverse the exhaust plenum.

   b) In the event that the main dampers fail to close as stipulated in Specification 3.7.1.2(a), the auxiliary intake and exhaust dampers shall close within ten seconds.

   c) A radiation monitor that samples the stack effluent shall be operating. This monitor shall indicate and alarm in the reactor control room.

d) The tritium concentration in the stack effluent shall be measured so as to provide the information required for reporting pursuant to Specification 7.7.1.8.

3. An installed instrument capable of detecting fission products shall be used to monitor the effluent in the purge gas that is drawn through the space between the reactor top lid and the surface of the primary coolant. Portable instruments, surveys, or analyses may be substituted for the installed monitor for periods of one week or until the next scheduled outage in cases where the MITR-II is scheduled to continuously operate. If the preceding is done, then the portable instrument shall be read and/or the survey or analysis performed at least daily. If there are elevated readings on the plenum effluent monitors, then the frequency shall be at least every eight hours.

4. Whenever the reactor floor is occupied, there shall be at least one area radiation monitor capable of warning personnel on the reactor floor of gamma radiation levels. If any area monitor is inoperable and work is to be done in that area, portable instruments shall be used to survey radiation in that area.

5. Whenever secondary coolant is flowing through the D_2O heat exchangers to the cooling tower the following shall be provided:

   a) The secondary water shall be sampled for tritium content either daily by manual means or continuously by an in-line monitor, and

   b) The levels of the primary storage tank, reflector dump tank, and the fission converter tank shall be monitored, either by low level alarms in the control room, or by hourly readings of the tank sight glasses or local gauge.

6. Whenever the reactor is operating with secondary coolant circulating between the containment building and the cooling tower, a secondary water monitor which indicates and alarms in the control room shall be operating.

7. At least one environmental monitor at the site and one within approximately one quarter mile of the site shall be used to verify compliance with environmental dose limits. These devices may be real-time monitoring or passive-monitoring. They shall be capable of detecting the radiation from the facility or radiation from effluent releases from the facility.

8. Setpoints for the required radiation monitors shall be as listed in Table 3.7.1-1.

Basis

The radiation monitoring system is described in Section 7.7 of the SAR. The MITR-II has three continuous air monitors, one on every level of the building. Any one of the three, or a portable device, can perform the required function.

If the containment is not isolated, an engineered safety feature in the form of an interlock between any one of the four plenum effluent monitors (two gas and two particulate) will cause the main ventilation isolation dampers to close. If these fail to close, the auxiliary ones will do so. The main ones will close before the release can leave the building. The auxiliary will close within 10 seconds. The effluents that would be released during even the maximum hypothetical accident during ten seconds are insignificant.

There are five possible stack monitors (2 gaseous, 2 particulate, and 1 area). Any one of the five fulfills the required function.

Tritium may be sampled by means of bubblers (one in the stack base and one in the exhaust plenum) or by means of special samples as described in Section 11.1.4.2 of the SAR.

The air purge that is drawn through the space above the primary coolant and below the reactor top lid is continuously monitored for fission product activity as described in Section 9.1.5.2 of the SAR. This detection method provides notice of any incipient fuel clad failures.

Area monitors are located throughout the containment building. At a minimum, one is required to be operable on the main reactor floor when it is occupied by experimenters. However, if any one of these monitors is inoperable, portable instruments will be substituted.

Monitoring for heavy water leakage into the secondary coolant is based on three independent measurements. These are:

a) The secondary water monitor is a gamma-sensitive scintillation detector. It cannot detect tritium but is sensitive to N^{16} and F^{17} which are also present in heavy water when the reactor is operating.

b) Either daily or continuous sampling of the secondary water will allow detection of very small leaks.

c) Because of the nature of the primary and reflector systems, any loss of coolant inventory will be reflected by a decrease in the level in either the primary storage tank or the reflector dump tank. Loss of fission converter coolant inventory will be reflected by a decrease in the level of the fission converter tank.

The secondary water monitors will also detect primary leakage.

The environmental monitors are used to verify that the potential maximum dose, annual or other, in the unrestricted environment is within the values analyzed in the SAR.

Action guidelines are chosen to alert the operator that attention to the condition is warranted. Guidelines are not absolute limits and operation in excess of these guidelines will not necessarily result in exceeding 10 CFR 20 limits which are based on annual averaged quantities.

Table 3.7.1-1

Required Radiation Monitors

Channel		Minimum Number	Alarm	Interlock/Action	Action Guideline (8)	Notes
Area		1 (reactor floor)	Alarm (local and control room)	None	5 mrem/h	(1) (2)
Sewer		1	Alarm (control room)	De-energize sewer pump	MAC (9)	(3)
Secondary Coolant		1	Alarm (control room)	None	1E-3 µCi/ml	(4)
Core Purge (off-gas)		1	Alarm (control room)	De-energize blower Isolates core purge	3x background (100k cpm)	(5)
Stack	Gaseous	1	Alarm (control room)	None	4E-4 µCi/ml	(6)
	Particulate		Alarm (control room)	None	4E-4 µCi/ml steady state or 2E-7 µCi/ml in 20 min rate of rise	
	Area		Alarm (control room)	None	N/A	(7)
Plenum	Gaseous	1	Alarm (control room)	Containment isolation	1.4E-4 µCi/ml	(6)
	Particulate		Alarm (control room)	Containment isolation	6E-4 µCi/ml steady state or 2E-7 µCi/ml in 30 min rate of rise	
Building Airborne Radioactivity	Gaseous	1	Alarm (local)	None	1 DAC	
	Particulate		None	None	0.3 DAC	

Notes:

(1) Set point is established at 5 mrem/h or 5 mrem/h above background dependent upon the normal radiation levels present in the vicinity of the ARM. In the event of special operations or temporary conditions, set points may be established different from the specification for the condition and returned to normal operating set point conditions upon termination of special operations or the temporary conditions.

(2) In the event that the channel is not operable, then qualified individuals may use radiation protection instrumentation in the work area, or use a portable unit with a local alarm capability to warn the workers, or through the issuance of electronic (alarming) pocket dosimeters.

(3) For discharge to sewer.

(4) Not required if reactor is secured and secondary coolant to D_2O heat exchanger(s) is secured.

(5) May be exceeded provided conditions of Specification 3.1.6 are met; portable instruments, surveys, or analyses may be substituted for one week or until the next scheduled outage.

(6) Particulate set point is established based on steady state values or rate of rise. Gaseous and particulate set points incorporate dilution factors and may be exceeded provided monthly average does not exceed effluent concentration limits.

(7) Alarm set point based on established emergency plan emergency action levels.

(8) Action guidelines are chosen to alert operator that attention to the condition is warranted. Guidelines are not absolute limits and operation in excess of these guidelines will not necessarily result in exceeding 10 CFR 20 limits which are based on annual averaged quantities.

(9) MAC is monthly average concentrations.

3.7.2 Effluents

Applicability

This specification applies to the radioactive effluents that are released from the reactor site.

Objectivity

To ensure that the release of radioactive effluents to the environment is within the limits of 10 CFR Part 20.

Specification

1. The release of radioactive effluents from the reactor site will comply with all provisions of Part 20, Title 10, Code of Federal Regulations with the following consideration. A dilution factor of 50,000 shall be applicable to the concentration of airborne effluents released from the stack. For particulates and iodines with half lives greater than eight days in each case, an effective dose scaling factor of 1,200 is applied to the dilution.

2. On indication of ≥ 1 μCi/liter of tritium in the secondary coolant water, the cooling tower spray shall be shut down, the secondary system water discharge shall be stopped, and the D_2O reflector heat exchangers shall be isolated until tritium leakage into the secondary has been controlled.

Basis

Concentrations of radioactive gases from the MITR-II stack will be maintained as low as reasonably achievable. Because of atmospheric diffusion and variation in wind direction, the yearly average concentration of radioactive materials in all unrestricted areas which could be occupied by individuals will be much less than the limits established in 10 CFR Part 20.

The use of a maximum dilution factor of 50,000 for radioactive effluents that are discharged from the MITR-II ventilation stack will still maintain all possible yearly averaged concentrations to a small fraction of the 10 CFR 20 limits in any occupied area. This dilution factor is applicable to all gaseous effluents with the exception of radioactive isotopes that are subject to environmental reconcentration such as Iodine-131. The dilution factor was calculated using the CAP88-PC code for straight-line Gaussian diffusion [3.7.2-1]. This gave a value of $>1x10^6$ at 900 m which was the point of maximum dose to an individual. Calculations made in conjunction with the MITR-II startup allowed for the effect of nearby buildings. These and subsequent calculations gave a dilution factor of at least 50,000. Accordingly, a factor of 50,000 is selected. An effective dose scaling factor of 1,200 is applied for radioactive iodines and particulates with half-lives greater than eight days to account for differences in dose pathways and dose conversion ratios. This factor is determined by estimating the dose from all pathways for Iodine-131 compared to noble gases for a unit release, based upon values generated using the CAP88-PC code. This factor ensures that, if the MITR-II stack releases of iodines and particulates with half-lives greater than eight days are kept within the 10 CFR 20 limits at the nearest point of public occupancy, the potential for radiation doses after dilution will be a small fraction of the 10 CFR 20 limits.

Liquid waste is discharged to the municipal sanitary sewer systems from two waste storage tanks and from the cooling tower. Radionuclide concentration limits set on the monitoring and sampling systems are such that conformity with the limitations specified in 10 CFR 20 is ensured.

The average discharge of water from the secondary coolant system of the reactor to the sanitary sewer system is approximately 13,000 gallons per operating day. This is diluted by an approximately equal volume of water from adjacent MIT buildings. The sewerage from the site enters the Cambridge sewer system where it is further diluted by discharge by the rest of MIT (at about 1 MGD) and by an unmeasured amount of storm drainage. The Cambridge sewerage enters a Metropolitan District Commission trunk sewer line. The estimated discharge rate from this line, based upon the permitted discharge rate, is 436 MGD [3.7.2-2].

Thus, the reactor effluent is diluted by a factor of about $4.3 \times 10^8 / 1.3 \times 10^4 = 30{,}000$ at the ultimate point of discharge from the sewer system. The proposed limit of 1 μCi/liter in the cooling tower water ensures concentrations at the point of discharge from the sewer system will be well below the limit of 10 CFR 20.

References

3.7.2-1 Parks, B., "Mathematical Models in CAP88-PC," U.S. Department of Energy, June 1997.

3.7.2-2 Massachusetts Water Resources Authority, "Monthly Compliance and Progress Report," June 1999.

3.7.3 Reactor Floor Hot Cells

Applicability

This specification applies to the reactor floor hot cells.

Objective

To ensure that the hot cells are utilized as described in the SAR.

Specifications

Whenever the hot cells are used the following shall be provided:

a) No experiments shall be conducted in the hot cells and on-going experiments shall be terminated when the containment building and/or hot cell ventilation systems are shut down.

b) The hot cell blower shall be interlocked with the containment building ventilation such that it does not operate unless the containment ventilation is operating.

c) In event of a fire in the hot cell, the hot cell blower shall be automatically shut off and an alarm shall sound in the hot cell area and in the control room.

d) A visual alarm in the hot cell area shall activate when the containment building exhaust system is shutdown.

Basis

The basis of this specification is given in Sections 10.2.8 and 10.3.2.7 of the SAR. When the containment ventilation system is shutdown, the exhaust for the ventilation gases and particulates from the hot cell is lost. Accordingly, experiments in the hot cells should not be initiated and those in progress should be terminated until the containment ventilation system is again operable.

In event of a fire in the hot cells, the hot cell blower that exhausts the atmosphere from the hot cells shall be shutdown. When the containment building exhaust system is shutdown, it is necessary to alert any operators involved in the hot cell operations that those operations must be terminated until the containment blowers are once more operable. Accordingly, visual alarms are located in the hot cell area.

3.7.4 Byproduct Material

Applicability

This specification applies to the receipt of byproduct material on the Reactor Operating License.

Objective

To establish criteria for the receipt of byproduct material on the Reactor Operating License.

Specification

1. Byproduct material will be limited to the following:

    a) Atomic numbers 3 through 83 in solid form for the purpose of materials studies not to exceed 100,000 Curies total inventory, 1000 Curies per specimen and not to exceed 100 rem/h at one meter from an unshielded source.

    b) Atomic numbers 3 through 83 in any form for the purpose of calibration, characterization, and detection for radiation protection purposes not to exceed 1000 Curies.

2. Byproduct material received under this provision may be irradiated provided that the irradiations comply with Specification No. 6.1, 'General Experiment Criteria'.

Basis

This authorization allows receipt of byproduct material for use in radiation damage studies. The material may be reirradiated subject to the provisions of Specification No. 6.1.

4. SURVEILLANCE REQUIREMENTS

4.1 Reactor Core Parameters

Applicability

This specification applies to surveillance of reactor core parameters.

Objective

To ensure that reactor core parameters are as specified in the analyses contained in the SAR.

Specification

1. Excess Reactivity: The operability of the subcritical limit interlock shall be verified at least annually.

2. Shutdown Margin: The shutdown margin shall be determined at least annually and after changes in core configuration, refuelings, the insertion/removal of in-core experimental assemblies, and control device (shim blade and/or regulating rod) changeouts.

3. Core Configuration: Refer to Specification 3.1.4.4(c).

4. Reactivity Coefficients: Refer to Specification 3.1.5. In addition, reactivity coefficients will be measured if analysis of a proposed change of core configuration or fuel type indicates that there could be a significant change in the magnitude of the coefficient in question.

5. Fuel Parameters:

a) Fuel elements shall be inspected annually in accordance with Specification 3.1.6.2(a) and 3.1.6.2(b).

b) The fission density limit (Specification 3.1.6.3) shall be calculated for all in-core elements whenever the reactor is refueled.

c) Self-protection of fuel in the spent fuel storage pool shall be verified by measurement at least biennially if any element has been discharged from the core for more than 10 years. Measurements which show that the worst-case elements (those with the least burnup and longest time since discharge) are self-protecting shall satisfy this requirement.

d) The measurement frequency required by Specification 4.1.5(c) shall be changed to annual if any element is within a factor of two of the self-protection limit.

Basis

The subcritical limit interlock limits the amount of excess reactivity by requiring that the reactor not attain criticality unless all shim blades are withdrawn by at least five inches. This physical interlock is preferable to an administrative limit. Shutdown margin is determined and documented in writing any time there is a change of core configuration, refueling, insertion/removal of an in-core sample assembly, or control device changeout.

Compliance with thermal-hydraulic limits is verified for every change of core configuration. This includes refuelings.

Reactivity coefficients were measured during the initial startup of the MITR-II. Coefficients have been periodically remeasured either as teaching exercise or as part of the program for the evaluation of digital control strategies. No significant changes have been noted. The commitment is therefore to investigate any change should one be observed and to remeasure if a change in core design or fuel could cause a significant change.

The procedure for performing fuel inspections is noted in the basis of Specification 3.1.6.

The fission density limit is calculated as part of the MITR-II fuel management program for every element. It is updated at every refueling for those elements that are in-core.

Self-protection of the fuel is ensured for at least a year by operating an element in-core for about a week. Elements are in-core typically for a three to five year period. Hence, there is no need to measure dose rates. After sixty days of cooling, radiation levels exceed 10,000 rem/h unshielded at one meter. Measurements of dose rates on elements in the spent fuel storage pool show that they remain self-protecting for at least 10 years [4.1-1]. No MITR-II element has ever dropped below the self-protection limit during the time that elapsed between its discharge from the core and its return to the U. S. Department of Energy.

<u>References</u>

4.1-1 File Memo (Fuel Self-Protection, February 1997).

4.2 Reactor Control and Safety Systems

Applicability

This specification applies to the surveillance of reactor control and safety systems. The surveillance requirements in specifications #1, #6a, and #6b of this section can be deferred during periods of reactor shutdown until the next planned period of reactor operation, in which case they shall be performed as soon as practicable when reactor operation resumes.

Objective

To ensure the reliability of the reactor control and safety systems.

Specification

1. Reactivity Worth of Control Devices: The integral and differential worths of the six shim blades and of the regulating rod shall be measured at least annually. Either calculations of the expected change or measurements shall be made upon changeout of an absorber and upon changes in core configuration that involve a new type of fuel or a change in the total number of non-fueled positions.

2. Rod Withdrawal and Insertion Speed: The withdrawal and insertion speed of each shim blade and the regulating rod shall be verified annually.

3. Scram Times: The scram time of each shim blade shall be verified annually or whenever any work has been done on either the shim blade, its electromagnet, or its associated drive. For purposes of this check, the scram time shall be measured from the full-out position to the 80% inserted position of the shim blade.

4. Scram and Power Measuring Channels: The instruments or channels listed in Table 4.2-1 shall be tested at least quarterly and each time before startup of the

reactor if the reactor has been in a secured condition or if the instrument or channel has been repaired or de-energized. Calibration of these instruments or channels (except those such as scram pushbuttons that do not require calibration) shall be done at least annually.

5. Channel Tests: Channel tests of the instruments or channels listed in Table 4.2-1 shall be performed if the channel or instrument has been modified or repaired.

6. The following instruments shall be calibrated and trip points verified when initially installed, any time a significant change in indication is noted, and at least annually:

   a) Period
   b) Neutron Flux Level
   c) Primary Coolant Outlet Temperature
   d) Core Tank Level
   e) Reflector Tank Level
   f) Primary Coolant Flow
   g) D_2O Reflector Flow
   h) Shield Coolant Flow

7. Thermal Power: The signals used to compute thermal power shall be calibrated at least annually.

Table 4.2-1

Surveillance of Scram and Power Measuring Channels

Instrument or Channel		Channel Test to Verify
1	Period[(1)]	Scram
2.	Neutron Flux Level[(1)]	Scram
3.	Primary Coolant Outlet Temperature	Scram
4.	Core Tank Level	Scram
5.	Reflector Tank Level	Scram
6.	D_2O Dump Valve Switch	Scram and Reflector Dump
7.	Air-Operator D_2O Dump Valve Switch	Reflector Dump
8.	Manual Major Scram	Magnet Cut-off, Reflector Dump, and Ventilation Trip
9.	Manual Minor Scram	Magnet Cut-Off
10.	Experiment Shutdown	As Specified in Experiment Approval
11.	Primary Coolant Flow[(2)]	Scram
12	D_2O Reflector Flow[(2)]	Scram
13.	Shield Coolant Flow[(2)]	Scram
14.	Fission Converter	As specified in Fission Converter TS 6.6.3
15.	Nuclear Safety Channel Low Count Rate[(1)]	Scram
16.	Nuclear Safety Channel in Test[(1)]	Scram
17.	Nuclear Safety Channel Fault[(1)]	Scram
18.	Hold-Down Grid Unlatched	Scram
19.	Reactor Remote Shutdown(s)	Scram from Medical Facilities and Utility Room

(1) Reactor scrams when two trips in any combination are present simultaneously from any two of the four nuclear safety channels.

(2) Not required for startup in natural convection cooling mode.

8. Heat Balance: The signal from the linear power channel shall be checked against a heat balance calculation at least monthly, for any month that the reactor is operated above 1 MW continuously for at least 48 hours.

9. Control Device Inspection: Control devices shall be inspected annually as follows:

   a) Shim blade absorbers shall be checked visually.

   b) Shim blade electromagnets shall be checked both visually and by measuring the resistance of the coils.

   c) Shim blade and regulating rod drives shall be monitored for proper operation.

10. Control System Interlocks: A channel test of the following interlocks and scram shall be performed at least annually:

   a) Withdraw Permit Interlock,

   b) Subcritical Limit – Shim Blades Interlock,

   c) No Overflow Reflector Startup Interlock, and

   d) Low Level D_2O Reflector Scram.

Basis

The MITR-II has observed the criteria given in Specification 4.2.1 for determination of control device reactivity worths and found it to be adequate. Measurements of the integral and differential worths are required annually. Because such measurements require operation of the reactor, they are deferrable until the reactor is operating. Measurements following changeouts of absorbers and change of core configuration are desirable. However, such measurements are very time consuming. Moreover, sufficient experience exists with such changes that their effect on integral and differential reactivity worths can be predicted with reasonable accuracy. Accordingly, normal MITR-II practice is to do a complete set of measurements following replacement of all absorber sections rather than to do measurements as each is replaced. (Note: It requires several days to replace one absorber and the entire process is usually done over an interval of several months.) Estimates of the change of worth are used pending the measurement. Estimates, not measurements, are normally used for changes of core configuration.

The insertion and withdrawal speed of the control devices is fixed by the motor and drive design as discussed in Section 4.2.2 of the SAR. These speeds are verified annually.

Scram time is as defined by Specifications 1.3.37 and 3.2.1. It is verified at least annually and whenever maintenance has been performed that could affect it.

The instruments and channels listed in Table 4.2-1 correspond to those in Table 3.2.3-1, "Required Safety Channels" with the exception that surveillance of the building overpressure and gasket deflated scrams is addressed elsewhere (Specification 4.4).

Some of the calibration procedures for the wide-range neutron flux monitors that calculate period and neutron flux level require operation of the reactor. These are therefore deferrable until the reactor is operating.

The thermal power indication is calibrated at least annually and the signal from the linear power channel is compared against a heat balance at least monthly for any month that the reactor is operated above 1 MW. These actions are done under conditions of thermal equilibrium which, because of the MITR-II's heat capacity (especially that of the graphite reflector), occurs after 48 hours of steady-state operation.

Control devices are inspected at least annually. The inspection focuses on those components that are important to safety. Those include the absorber sections (Section 16.3.1.5 of the SAR) and electromagnets (Section 16.3.1.4(d) of the SAR). The status of the shim blade and regulating rod drives can be deduced from external observations such as the measurement of blade and regulating rod insertion/withdrawal speeds (Specification 4.2.2). Internal inspections require lowering of the core tank level and removal of the drive. These are usually done whenever an absorber is changed out. As described in Section 16.3.1.5 of the SAR, this is normally done every 125,000 MWH. A prespecified frequency for an internal inspection would involve serious ALARA issues.

4.3 Coolant Systems

Applicability

This specification applies to the surveillance of the reactor coolant systems.

Objective

To ensure that reactor coolant systems are maintained as specified in the analyses in the SAR.

Specifications

1. Each emergency core cooling system (ECCS) shall be tested at least annually. The minimum flow through each ECCS spray nozzle system shall be 10 gpm. The test shall include verification of operability of the manual valves to the city water supply lines and the core spray nozzles.

2. If an In-Core Sample Assembly (ICSA) of a type not previously evaluated is to be installed, verification that the ECCS spray nozzles are positioned so that each fuel element receives at least 20% of the average flow shall be determined by measurement using an ex-core mock-up if the ICSA has the potential to obstruct ECCS spray.

3. In-service inspections of primary coolant system core components shall be performed annually.

4. Analysis of the primary, D_2O, and secondary coolant for radioactivity shall be as follows:

Analyses	Frequency
Primary gross activity	Weekly during any week that the reactor is operating above 1 MW continuously for at least 24 hours and at least quarterly.
Primary isotopic identification	Quarterly
D_2O gross activity	Quarterly
D_2O tritium	Quarterly
Secondary gross activity	Daily any day that the reactor is operating or that secondary flow is supplied to a D_2O exchanger.
Secondary tritium	Daily any day that the reactor is operating or that secondary flow is supplied to a D_2O exchanger.

5. The pH and conductivity of the primary coolant shall be measured at least quarterly. Whenever the reactor is operating, primary conductivity shall be verified at least weekly and if it is out of specification, pH and chloride ion concentration shall also be measured weekly.

6. The pH and conductivity of the fuel storage pool water shall be measured at least quarterly.

<u>Basis</u>

The test of each emergency cooling system will consist of opening the manual valve in the city water supply line to ensure proper operation of the valve and of spraying water into the core through the core spray nozzle. During the test, the normal flow path will not be used in order to prevent adding city water to the primary reactor coolant. The city water will be diverted to the liquid effluent system without discharging through the spray nozzle while primary coolant water will be used to test the spray nozzle.

In-core sample assemblies may contain extensions that serve as conduits for the experiment coolant, sensors, or heater cables. These conduits may obstruct ECCS nozzle spray. If an ICSA of a type not previously evaluated is to be installed and if it has the potential to obstruct ECCS nozzle spray, then an ex-core mock-up is used to verify that each fuel element will receive at least 20% of the average ECCS spray.

In-service inspections of primary coolant system core components are performed at the same time as the fuel inspection required by Specifications 3.1.6 and 4.1.5(a).

The minimum frequency for pH and conductivity sampling is quarterly. However, primary samples are taken prior to most startups and conductivity is monitored on-line. For pure water, pH and conductivity vary linearly with each other. Therefore, the reading of the conductivity instrument is checked at least weekly and, if the conductivity is out of specification, the pH is measured. If the conductivity is in specification, the pH will also be in specification.

4.4 Containment Surveillance

Applicability

This specification applies to the surveillance of the reactor building containment and its associated systems including the vacuum breakers, pressure relief system, and associated scrams.

Objective

To ensure that building integrity is as specified in the SAR.

Specification

1. An integral air leak rate test of the reactor building containment shall be performed biennially. Each test shall be performed at a test pressure greater than 1 psig and less than 2 psig with the blow-off leg set at below 2 psig.

2. Leak tests of individual penetrations shall be performed between integral tests when either new penetrations or repairs of existing penetrations are made. The sum of the results of the last integral building leak rate test and any increase in the penetration leakage since the integral test must satisfy Specification 3.4.3.

3. The main and auxiliary intake and exhaust ventilation isolation dampers shall be inspected annually.

4. A test of the proper functioning of the independent vacuum relief breakers shall be performed annually.

5. A test of the charcoal filters in the pressure relief system shall be performed annually to determine their efficiency for the removal of the elemental iodine. The filters shall be replaced if the efficiency is 95% or less.

6. Flow shall be established through the charcoal filters in the pressure relief system for 15 minutes or longer, at least monthly and each time before startup of the reactor if the reactor has been shut down more than 24 hours.

7. The following scrams shall be tested at least quarterly or if the scram has been modified or repaired.

Scram	**Test**
Building Overpressure	Channel Test
Main Personnel Lock Gaskets Deflated	Channel Test
Basement Personnel Lock Gaskets Deflated	Channel Test

Basis

The reactor containment building has been tested annually for leakage from 1958 to 1999. On only one occasion has the facility failed to pass the leak rate test. In that one case, the margin by which the facility failed was very small and was caused by deterioration of the rubber gaskets on the intake and exhaust ventilation isolation dampers. Because each of these two dampers is redundant (main and auxiliary), the building could still have been maintained at the test pressure within the permissible leak rate in the event of an emergency. Inspection of the containment design indicates that the most probable points of excessive leakage are the isolation dampers, which has been confirmed by the containment testing history as stated above. Other seals on the reactor building are either welded or duplicated. Failure of these seals is therefore much less probable than damper gasket failure.

In view of this conclusion, the ventilation isolation dampers will be visually inspected annually. An integrated leak rate will be conducted biennially. The test is conducted in two parts in order to check redundant closure devices individually. The results of these tests are corrected for temperature, pressure, and humidity changes, and an error analysis is also made to estimate the limits of uncertainty in the measurements.

New or repaired penetrations are tested as necessary using special procedures that are prepared for the particular penetration.

The vacuum breakers are tested under pressure as an integral part of the containment leakage test. It is necessary also to test them under vacuum to ensure that they will open properly.

The building pressure relief charcoal filters will be tested by methods described in Section 7.5.2 of Reference 4.4-1 or the equivalent. Successful experience with the system over the past twenty-five years justifies the annual frequency of testing.

Air flow is established periodically through the charcoal filters in order to maintain the charcoal activated so that it will remove iodine. This is done in accordance with the vendor's recommendations.

Scram channels associated with the containment building are tested quarterly. Experience over the past twenty-five years (1975-1999) has shown this frequency to be more than adequate.

<u>Reference</u>

4.4-1 Burschsted, C. A. and A. B. Fuller, "Design, Construction, and Testing of High-Efficiency Air Filtration systems for Nuclear Application," ORNL-NSIC-65, January, 1970.

4.5 Ventilation Systems

Applicability

This specification is applicable to surveillance of the containment building ventilation system.

Objective

To ensure that the containment building ventilation system functions as described in the SAR.

Specification

1. The flow rate through the ventilation systems exhaust stack shall be measured at least annually.

2. The following interlocks shall be tested at least quarterly and each time before startup of the reactor if the reactor has been shut down for more than 24 hours or if the equipment/circuitry involved has been repaired.

Interlock	**Surveillance**
Building ΔP - Reactor Start	Channel Test
Plenum Effluent Monitor - Main Damper Trip	Channel Test
Main - Auxiliary Damper Interlock	Channel Test
Ventilation – Hot Cell Blower Interlock	Channel Test

3. The Building ΔP - Reactor Start interlock shall be calibrated annually.

4. The ΔP across the filters in the ventilation exhaust shall be monitored and the filters replaced when the ΔP exceeds the manufacturer's recommendations.

Basis

Experience from both the MITR-I and MITR-II shows that an annual measurement of ventilation flow rate is sufficient to detect trends. The interlocks listed under Specification 4.5.2 are tested in the same manner as scram channels. Experience has shown that annual calibration of the "Building ΔP - Reactor Start" interlock is adequate. The ΔP across the filters is monitored by permanently installed manometers. These allow the ΔP to be monitored for trends.

4.6 Emergency Electrical Power Systems

Applicability

This specification applies to the surveillance of the emergency electrical power supply.

Objective

To ensure the reliability of the reactor control and safety systems.

Specification

1. The temperature of the negative terminal on each battery cell shall be measured quarterly.

2. The voltage of each battery cell shall be measured semi-annually.

3. The connector resistance at each battery cell shall be measured annually.

4. A discharge test shall be performed once every two years

5. Operability of the inverter motor-generator set and associated switches shall be verified annually. Performance of a discharge test satisfies this requirement.

Basis

The emergency electrical power system consists of batteries, an inverter motor-generator set, and the switches necessary to tie into the normal electrical distribution system.

Voltage measurements of individual battery cells are the accepted method of ensuring that the batteries are in satisfactory condition. In addition, periodic discharge tests are performed to detect deterioration of cells. To ensure the operability of the inverter motor-generator set, the generator and associated switches will be operationally tested.

The frequency of these component tests is based on manufacturer recommendations and on standard practice as recommended in ANSI/ANS-15.1-2007. Specific gravity measurements are not needed for lead-acid battery cells that use no free-flowing electrolyte. The discharge test frequency is biennial rather than every five years.

4.7 Radiation Monitoring Systems and Effluents

4.7.1 Radiation Monitoring Systems

Applicability

This specification applies to the surveillance of the radiation monitoring systems.

Objective

To ensure that radiation monitoring systems are maintained as specified by the SAR.

Specifications

1. A channel check shall be made of the area and effluent (stack and secondary coolant) radiation monitors on any day that the reactor is operating above 250 kW for at least 12 hours.

2. The following radiation monitors shall be tested at least monthly and each time before startup if the reactor has been shut down for more than 24 hours or if the instrument has been repaired or de-energized.

Monitor	**Test**
a) Area Radiation	Channel Test Using a Source
b) Plenum Gas and Particulate	Channel Test Using a Pulse
c) Stack Gas and Particulate	Channel Test Using a Pulse
d) Secondary Coolant	Channel Test Using a Pulse
e) Core Purge Monitor	Channel Test Using a Pulse

3. The following radiation monitors shall be tested quarterly:

Monitor	**Test**
a) Plenum Gas and Particulate	Channel Test Using a Source
b) Stack Gas and Particulate	Channel Test Using a Source
c) Secondary Coolant	Channel Test Using a Source
d) Core Purge	Channel Test Using a Source
e) Sewer	Channel Test Using a Source

4. The radiation monitors listed in Specification 4.7.1.3 and the area radiation monitors shall be calibrated and the trip points verified when initially installed and annually thereafter.

5. The continuous air monitor shall be calibrated and the trip point verified when initially installed and annually thereafter.

6. The in-line tritium monitor shall be calibrated and the trip point verified when initially installed and annually thereafter.

Basis

The channel check provides a qualitative verification of the performance of the radiation monitors. The channel tests verify operability by the introduction of a test signal. The calibration provides a complete verification of the performance of the instrument.

The annual calibration of the in-line tritium monitor is per the manufacturer's recommendation. The monitor is equipped with a self-checking circuit that will provide warning of any failure during routine use.

4.7.2 Effluents

Applicability

This specification applies to the surveillance of the quantities of radioactive effluents released to the environment.

Objective

To ensure that releases to the environment are in compliance with the requirements of 10 CFR 20.

Specification

1. Continuous monitoring of effluents and collection of information shall be performed as required for reporting as specified within Specification 7.7.1.8.

2. Comparison of the information collected to other methods such as environmental monitoring to verify the validity of the data shall be used as necessary.

Basis

This information is documented in the facility's annual report.

5. DESIGN FEATURES

5.1 Site and Facility Description

Applicability

This specification applies to the reactor site and facility.

Objective

To ensure that features of the site and facility which, if altered, would significantly affect safety are specified.

Specification

1. The reactor facility is located at 138 Albany Street on the MIT Campus in the City of Cambridge, Massachusetts.

2. The distance to the nearest point of normal public occupancy is at least 68 feet.

3. The restricted area shall include the reactor containment building, the adjoining one-story building, and the fenced-in area in which the cooling tower, ventilation exhaust stack, and liquid waste storage tanks are located.

4. The height of the ventilation exhaust stack is at least 150 feet.

5. The volume of the reactor containment building is approximately 200,000 cubic feet.

Basis

The site location is specified to ensure ownership of the site by the licensee. The reactor site is described in Section 2.1.1 of the SAR. The closest point of public access is a railroad track that runs parallel to the south side of the restricted area. However, that location is normally not occupied. The nearest point of normal public occupancy is the sidewalk on the northerly side of the site.

Access to the restricted area is limited to authorized personnel only.

The height of the ventilation exhaust stack and the volume of the containment building are design features that bear on radiological safety. The figures specified are the ones used in the SAR for effluent calculations.

5.2 Primary Coolant System

Applicability

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

Objective

To ensure compatibility of the primary coolant system with the safety analysis.

Specification

The reactor coolant system shall consist of a reactor vessel, a single cooling loop containing one or more heat exchangers, and appropriate pumps and valves. All materials, including those of the reactor vessel, which are in contact with the primary coolant (H_2O) shall be aluminum alloys, stainless steel, or titanium alloys, except for small non-corrosive components such as gaskets, filters, and valve diaphragms. The reactor vessel shall be designed in accordance with the ASME Code for Unfired Pressure Vessels. It shall be designed for a working pressure of 24 psig and 150 ˚F. Heat exchangers shall be designed for 75 psig and a temperature of 150 ˚F. The connecting piping shall be designed to withstand a 60 psig hydro test.

Basis

The reactor coolant system originally consisted of a single loop that contained two heat exchangers. It was subsequently modified to add a third heat exchanger although it was normally operated with only two heat exchangers on line. Core safety is unaffected by the number of heat exchangers provided that the required heat transfer surface area is available for heat removal and that primary coolant flow remains as required by Specification 3.2.3.

The materials of construction are primarily aluminum alloy, stainless steel, and titanium alloys, and are chemically compatible with the H_2O coolant. The design, temperature, and pressure of the reactor vessel and other primary system components provide adequate margins over operating temperatures and pressures. The reactor vessel was designed to Section VIII, 1968 edition, of the ASME Code for Unfired Pressure Vessels. Subsequent design changes will be made in accordance with the most recent edition of this code.

5.3 Reactor Core and Fuel

Applicability

This specification applies to the design of the reactor core and the fuel.

Objective

To ensure compatibility of the reactor core and fuel with the present safety analysis.

Specification

1. The reactor core may consist of up to 27 fuel elements approximately 2-3/8" on a side. The fuel shall be plates of uranium in the form of UAl_x with a maximum of 50 w/o uranium in the fuel matrix clad by a layer of aluminum metal that incorporates fins on the surface to enhance heat transfer. The fuel plates shall have a nominal clad thickness not less than 0.015 inches at the base of the groove between the fins, with local regions not less than 0.008 inches. The fuel loading shall be:

   34.0 + 0.2, -1.0 grams U-235 per plate
   510+3.0, -10.0 grams U-235 per element

2. Design of in-core sample assemblies shall conform to the following criteria:

   a) They shall be positively secured in the core to prevent movement during reactor operation.

   b) Materials of construction shall be radiation resistant and compatible with those used in the reactor core and primary coolant system.

   c) Sufficient cooling shall be provided to ensure structural integrity of the assembly and to preclude any boiling of the primary coolant.

   d) The size of the irradiation thimble shall be less than 16 square inches in cross section.

Basis

The thermal design analysis in the SAR and the power distributions on which the analysis is based assume fuel elements of the type specified in Specification 5.3.1. Any change in this design would require re-evaluation of the heat transfer and flow characteristics of the element.

The nominal clad thickness of 0.015 inches is based upon standard practice for MTR type elements with clad of similar thickness. Reference 5.3-1 states that the release of radioactive fission gas to the primary cooling water appears to be adequately prevented by cladding of uniform thickness >0.2 mm (>0.0079 inches). Fabrication specifications for MITR-II fuel provide for a nominal clad thickness of 0.015 inches at the base of the grooves. Fabrication tolerances and minor manufacturing deviations (e.g., scratches, indentations, etc.) may result in the clad thickness being reduced to not less than 0.008 inches in local regions. Reference 5.3-2 shows that a thick clad increases the delay time for heat removal in the event of a fast transient. Therefore, the clad should be as thin as possible while still remaining compatible with fission product retention requirements.

In-core sample assemblies which satisfy Specification 5.3.2(a) cannot be credibly ejected during operation and are therefore considered part of the reactor structure. Specifications 5.3.2(b) and 5.3.2(c) ensure the structural integrity of the assembly and prevent chemical interactions with the core and primary coolant system. Specification 5.3.2(d) limits the size of the irradiation area as required by 10 CFR 50.2.

References

5.3-1 Beeston, J. M., R. R. Hobbins, G. W. Gibson, and W. F. Francis, "Development and Irradiation Performance of Uranium Aluminide Fuels in Test Reactors," *Nuclear Technology*, Vol. 49, p. 136-149 (June 1980)

5.3-2 Thompson, T. J. and J. G. Beckerly (eds.) *The Technology of Nuclear Reactor Safety*, Vol. I, The MIT Press, Cambridge, Mass. (1964)

5.4 Fuel Element Storage

Applicability

This specification applies to the storage and handling of fuel elements.

Objective

To ensure that fuel elements which are not currently in the core will be stored safely.

Specification

1. Fuel elements shall be stored in accordance with the requirements of the MITR-II Security Plan.

2. Unirradiated fuel elements may be stored in any of the following locations, subject to the MITR-II Security Plan and to Specification 4 below:

   a) In the reactor core provided that the reactivity is below the shutdown margin given by Specification 3.1.2,

   b) In the cadmium-lined fuel storage ring that is attached to the flow shroud, and

   c) In the storage safe in the reactor containment building (fuel plates also).

3. Irradiated fuel elements may be stored in any of the following locations:

   a) In the reactor core provided the reactivity is below the shutdown margin given by Specification 3.1.2,

   b) In the cadmium-lined fuel storage ring attached to the flow shroud,

   c) In the spent fuel storage pool in the basement of the reactor building,

   d) In the fuel element transfer cask or other proper shield within the controlled area, and

   e) In the fission converter tank.

4. A maximum of one fuel element shall be moved in or out of the reactor core at a time. Not more than four of the MITR-II fuel elements or the equivalent of two fuel elements including loose plates (maximum of 15 loose fuel plates) shall be outside of the approved storage areas except during the processes of receiving or shipping fuel from the site in approved containers. In all cases of fuel element storage outside of the reactor core, the value of k_{eff} shall be less than 0.90. Records of fuel element transfers shall be maintained. Prior to transferring an irradiated element from the reactor vessel to the transfer cask, the element shall not have been operated in the core at a power level above 100 kW for at least four days.

Basis

One of the principal requirements in regard to storage of fuel elements is prevention of accidental criticality. The locations given in Specifications 5.4.2 and 5.4.3 provide for complete criticality control. The reactor itself is, of course, shielded and appropriate written procedures ensure that it is loaded properly. The fuel elements in the cadmium-lined storage ring in the core tank are neutronically isolated from the reactor core by the cadmium of each individual box and by the six control blades that surround the reactor core. The fresh fuel storage safe and the spent fuel storage pool both have carefully designed geometric arrays to ensure that criticality will not occur. The row of storage positions in the safe are separated by cadmium, and each fuel element box in the pool is lined with cadmium. The transfer cask can only hold one element at a time. Finally, the fission converter is designed as a subcritical facility.

Calculations have been made, by using the methods described in Chapter 4 of the SAR, which indicate that at least 8-1/3 MITR-II fuel elements with optimum spacing and optimum moderation are required for criticality. Similar calculations show that 72 fuel plates with 27 grams U-235 are required for criticality, giving a safe handling limit of about 30. The specification of no more than four elements outside of the designated storage areas ensures that no criticality will occur elsewhere. No calculations have been done for 34 gram plates, and so a conservative limit of 15 unassembled plates is established.

It has been calculated that fuel elements when stored in the locations specified in 2b, 2c, 3b, 3c, 3d, and 3e will have an effective multiplication (k_{eff}) factor of less than 0.9 under optimum conditions of water moderation [5.4-1, 5.4-2].

The chief additional problems with spent fuel are those of shielding personnel from the emitted fission product gamma rays and preventing melting from decay heat. The shielding requirement is met by utilizing a shielded transfer cask (item 3d) for movements and temporary storage and more permanent shielding as indicated in 3a, b, c, and e. The requirement to prevent melting is met by specifying that four days elapse between use of the fuel element in a core operating above 100 kW and removal of the element from the core tank. Calculations show that radiant heat transfer is adequate to remove the decay heat thereafter [5.4-3].

References

5.4-1 S. Dinsmore Report, "MITR Criticality Safety Evaluation of Storage Locations for 510 Gram Fuel Elements," MITR SR# O-80-13, 11 August 1980.

5.4-2 Fission Converter Safety Evaluation Report.

5.4-3 File Memo (Decay Heat Removal Using Radiant Heat Transfer, May 1999)

6. EXPERIMENTS

6.1 General Experiment Criteria

Applicability

This specification applies to experiments that use the reactor.

Objective

To ensure that experiments that use the reactor do not affect the safety of the reactor.

Specification

All experiments that are within the reactor or its surrounding structure shall conform to the following conditions:

1. Reactivity Effects

   The reactivity worth of experiments shall not exceed the values indicated in the following table:

	Single Experiment Worth	**Total Worth***
Movable	0.2% ΔK/K	0.5% ΔK/K
Non-secured	0.5% ΔK/K	1.0% ΔK/K
Total of the above	N/A	1.5% ΔK/K
Secured	1.8% ΔK/K*	N/A

*Total worths are to be determined by summing the absolute value of the reactivity worth of each single experiment.

2. Thermal-Hydraulic Effects

a) All experimental capsules shall be designed against failure from internal and external heating at the reactor power level or relevant process variable that corresponds to the Limiting Safety System Setting associated with that power level or process variable.

b) The outside surface temperature of a submerged experiment or capsule shall not cause nucleate boiling of the reactor coolant during operation of the reactor.

c) The insertion of an experiment into the core shall not cause a coolant flow redistribution that could negate the safety considerations implicit in the limiting safety system settings.

3. Chemical Effects

a) Metastable or other materials that could react to create a rapid pressure rise shall be encapsulated. The capsule shall be prototype-tested under experimental conditions to demonstrate that it can contain without failure an energy release equivalent to at least twice the material to be irradiated or at least twice the pressure that could be expected from any reaction of these materials. These tests must also include effects of any fragments which may be generated. If a change in experimental conditions could result in a greater potential for failure than design experimental conditions, the capsule shall also be tested under these changed conditions. In addition, the quantity of material should be limited such that if the maximum calculated energy release should occur, significant damage to the reactor core will not result, assuming that the material is not encapsulated.

b) No explosive materials (defined to include all materials that would constitute Class A, Class B, and Class C explosives as described in Title 49, Parts 172 and 173 of the Code of Federal Regulations) shall be placed in the reactor core or within the primary biological shield, which, if completely detonated, could cause any rearrangement or damage to the reactor core. Proposed quantities of explosive materials greater than the equivalent of 25 milligrams of TNT shall require a documented safety analysis and approval by the MIT Reactor Safeguards Committee. Capsule designs for explosive materials shall be prototype-tested to demonstrate that they can contain at least twice the pressure produced inside the capsule as a result of detonation of the material or the pressure produced by the detonation of twice the amount of material.

c) Corrosive materials that could affect or react with another material present in the reactor system shall be doubly encapsulated. If the material can adversely affect the reactor core or any of its component parts or auxiliary systems or the building containment to cause loss of function of the affected component or system, means shall be provided to monitor the integrity of the material container.

4. Radiolytic Decomposition

a) Compounds subject to radiolytic decomposition shall be irradiated in containers that can withstand the maximum gas pressure produced as result of the decomposition under irradiation including the effect of any temperature rise. This pressure shall be determined by previous experience or by testing as described in Specification 6.1.6.

b) Consideration shall also be given to any pressure buildup resulting from the decomposition of the sample container, such as might occur with a polyethylene vial.

c) Compounds subject to radiolytic decomposition may be irradiated in a capsule that is vented, provided that the vented release is less than 10.0% of the limits of 10 CFR 20 at any point of possible exposure. The total number of vented capsules shall be limited so that the limits of 10 CFR 20 at any point of possible exposure are not exceeded.

5. Experiment Scrams

Experiment scrams may be added for the protection of the experimental equipment and/or reactor components in the event of some malfunction. If malfunction of the experiment can adversely affect the reactor core or any of its component parts or auxiliary systems or the building containment to cause loss of function of the affected component or system, the experiment scrams shall be redundant.

6. Prototype Testing

Materials whose properties (composition, heating, radiolytic decomposition, etc.) are uncertain shall be prototype tested. These tests will be designed to give a stepwise approach to final operating conditions. The tests may either be stepwise time or flux irradiations with proper instrumentation to determine temperature, pressure, and radioactivity for each step as required.

7. Radioactive Releases

Experiments shall be designed so that malfunctions and normal operations are not predicted to result in exposures in excess of the limits of 10 CFR 20 to either onsite or offsite personnel or in releases of radioactivity in excess of the 10 CFR 20 annual average concentration limits.

Bases

Accidents resulting from the step insertion of reactivity are discussed in the SAR. The 0.2% ΔK/K limit for movable experiments corresponds to a 20-second period, one which can be easily controlled by the reactor operator with little effect on reactor power. The limiting value for a single non-secured experiment, 0.5% ΔK/K is set conservatively below the prompt critical value for reactivity insertion and below the minimum shutdown margin. The sum of the magnitudes of the static reactivity worths of all non-secured experiments, 1.0% ΔK/K, does not exceed the minimum shutdown margin. The total worth of all movable and non-secured experiments will not reduce the minimum shutdown margin as the shutdown margin is determined with all movable and non-secured experiments in their most positive reactive states. Finally, it was determined that, following a step increase of 1.8% ΔK/K, fuel plate temperatures would be below the clad melting temperature and significant core damage would not result.

Specifications 6.1.2 - 6.1.6 are intended to minimize the probability of experiment failure. Experiment capsules should be designed to withstand expected temperatures, pressures, chemical and radiochemical effects. The requirement for testing containers at twice the pressure or with twice the amount of explosive or metastable material to be irradiated provides a factor of two safety margin as allowance for experimental uncertainties. Table 6.1-1 gives a summary of the requirements for specimen irradiations for ease of review and classification of the specifications.

The radiological consequences of experiment malfunctions must be considered as stated in Specification 6.1.7. Consistent with the Commission's regulations, predicted onsite personnel exposures or offsite concentrations resulting from these malfunctions must not be in excess of those permitted by 10 CFR Part 20.

Table 6.1-1

Summary of Requirements for Specimen Irradiations

Requirement	Stable	Metastable	Explosive	Corrosive	Radiolytically Decomposible
Consideration of reactivity effects, induced activity, heating, and temperature distribution	x	x	x	x	x
Estimation of pressure buildup		x	x		x
Single encapsulation		x	x		x
Double encapsulation				x	
Container pressure test		x	x		
Other		d	a, d	b	c

a) Amounts above the equivalent of 25 mg TNT require safety analysis and approval of MITRSC.

b) If corrosion can cause loss of functions of the reactor core or any of its component parts, or auxiliary systems, or the building containment, integrity of container must be monitored during irradiation.

c) Container may be vented if release is less than 10.0% of 10 CFR 20.

d) Amounts limited such that reaction will not damage reactor core.

6.2 Vacant

6.3 Vacant

6.4 Closed-Loop Control Systems

Applicability

This specification applies to systems for the closed-loop control of the reactor exclusive of those automatic controllers covered by Specification 3.2.2. (Note: The reactivity restrictions contained in Specification 6.1.1 do not apply to experiments performed under this Specification 6.4.)

Objective

To ensure that the reactor can be safely shut down at any time.

Specification

1. Shim blades and/or the regulating rod may be connected to a closed-loop controller provided that the overall controller is designed so that the control of reactor power will always be feasible at either the desired termination point of any transient or at the maximum allowed operating power. Only one shim blade shall be withdrawn at a time.

2. Each proposed closed-loop controller shall require a documented safety analysis and approval by the MIT Reactor Safeguards Committee (MITRSC) or, if authorized by the MITRSC, by its Standing Subcommittee.

3. The nuclear safety system shall be separate from any closed-loop controller.

4. A period trip set at or longer than 20 seconds shall be operable whenever any closed-loop controller is in use. This trip shall transfer control to manual and sound an alarm.

5. The operability of the period trip shall be tested prior to use of any closed-loop controller during any week that a closed-loop controller is to be used.

Facility Specific Definitions

1. A reactor together with a specified control device is defined as constituting a system that is "feasible to control" if it is possible to transfer the system from a given power level and rate of change of power to a desired, steady-state power level without overshoot, or conversely, undershoot. (Note: If a deviation band is specified about the desired power level, then the terms "without overshoot" or "without undershoot" means that there will be no overshoot or undershoot beyond the permitted deviation.)

2. The word "separate" means that the output of an instrument used in the safety system will not be influenced by interaction with the control system. For example, a signal derived from an instrument that forms part of the safety system would not be transmitted to the control system unless first passed through an isolation device.

Basis

The basis of the specification is given in Section 10.3.2.8 of the SAR. Digital control of the MITR-II is permitted subject to either of two approaches. The first is to place a limitation on the reactivity worth of the control device that is associated with the controller. This is the traditional approach that is commonly associated with analog controllers. It is covered by Specification 3.2.2. The second approach is to design the controller so that it incorporates the concept of feasibility of control. The requirement that a closed-loop controller be designed so that control will always be feasible at the desired termination point of any transient ensures that there will be no power overshoots, or conversely, undershoots other than those permitted by specified deviation bands, if any. The basis for the approach is that the reactor period can be made rapidly infinite if the total reactivity, both that added directly by the control devices and that present indirectly from feedback effects, is maintained less than the maximum available rate of change of reactivity divided by the effective, multi-group decay parameter. Physically, if the reactivity is so constrained, then, by reversal of the direction of motion of the specified control device, it will be possible to negate the effect of the reactivity present and make the period infinite at any time during the transient. This condition, the absolute reactivity constraint, is unnecessarily restrictive. A less stringent constraint may be written that specifies that there be sufficient time available to eliminate whatever reactivity is present beyond the amount that can be immediately negated by reversal of direction of the designated control mechanism before the desired power level is attained. This condition is the sufficient reactivity constraint. This constraint's function is to review the decision of whatever control law is being used and, if necessary, override that decision. Provided that the net reactivity is always restricted to that permitted by the sufficient constraint, it should always be possible to halt a power increase before the desired termination point is attained by merely reversing the direction of travel of the control device. Therefore, adherence to this constraint means that no automatic control action should ever result in a challenge to the nuclear safety system. Additional information is given in References 6.4-1 and 6.4-2.

The requirement that each proposed closed-loop controller require a documented safety analysis and approval by the MITRSC or, if authorized by the MITRSC, by its Standing Subcommittee, ensures that the design of each controller will be carefully reviewed and that necessary off-line testing will be performed.

The requirement that the nuclear safety system be separate from that of any closed-loop controller means that the capability of that safety system to perform its intended function will not be compromised.

The existence of a trip that will transfer control to manual should the period become equal to or shorter than 20 seconds will provide a safety factor set more conservatively than the nuclear safety system. The signal used for this trip is separate from the nuclear safety system and is not processed by the closed-loop controller. This ensures that the capability of the trip signal to perform its intended function will not be compromised.

References

6.4-1 Bernard, J.A., Henry, A.F., and D.D. Lanning, "Application of the 'Reactivity Constraint Approach' to Automatic Reactor Control," *Nuclear Science and Engineering*, Vol. 98, No. 2, Feb. 1988, pp 87-95.

6.4-2 Bernard, J.A. and D.D. Lanning, "Considerations in the Design and Implementation of Control Laws for the Digital Operation of Research Reactors," *Nuclear Science and Engineering*, Vol. 110, No. 4, Apr. 1992, pp 425-444.

6.5 Generation of Medical Therapy Facility Beams for Human Therapy

Applicability

This specification applies solely to the generation of medical therapy facility beams for the treatment of human patients. It does not apply to any other use of the medical therapy facilities and/or their beams. Surveillances listed in this specification are only required if human therapy is planned for the interval of the surveillance. However, in the event of a hiatus in the scheduled performance of any given surveillance, that surveillance shall be performed prior to the initiation of human therapy during the interval in question.

Objective

To provide for the protection of the public health and safety by ensuring that patients are treated in accordance with the treatment plan established by the BNCT physician authorized user and that the ALARA principle is observed for all non-therapeutic radiation exposures.

Specification

1. Patients accepted for treatment shall have been referred by written directive from a BNCT physician authorized user from a medical center with an NRC or Agreement State medical use license that contains BNCT specific conditions and commitments for BNCT treatment on humans conducted at the Massachusetts Institute of Technology Research Reactor's Medical Therapy Facilities.

2. All medical treatments, including irradiations and analyses of the neutron capture agents in the patients, are the responsibility of the BNCT physician authorized user in charge of the therapy and the medical physicists from the

NRC-licensed or Agreement State-licensed medical center. The Massachusetts Institute of Technology is only responsible for providing current and accurate beam characteristic parameters to the medical use licensee and for delivery of the desired radiation fluence as requested in the written directive. Before the start of a therapy, both the certified medical physicist and the BNCT Principal Investigator, or designate, must agree that the therapy can be initiated. The BNCT physician authorized user is responsible for monitoring the therapy and for directing its termination. Because MIT is responsible for delivery of the prescribed fluence, the BNCT Principal Investigator, or designate, will under normal circumstances terminate the irradiation whenever the prescribed fluence is attained. However, a radiation therapy can also be terminated at any time if either the BNCT physician authorized user or the BNCT Principal Investigator, or designate, judges that the therapy should be terminated.

3. It shall be possible to initiate a minor scram of the reactor from a control panel located in each medical therapy facility area.

4. Access to each medical therapy facility shall be controlled by means of the shield door located at its entrance.

5. The following features and/or interlocks shall be operable:

   a) An interlock shall prevent opening of the shutters that control beam delivery unless the medical therapy facility's shield door is closed.

   b) The shutters that control beam delivery shall be interlocked to close automatically upon opening of the medical therapy facility's shield door.

c) Except for the fission converter mechanical shutter, the shutters that control beam delivery shall be designed to close automatically either upon failure of electric power, or upon reduced air pressure if the shutter is operated pneumatically. For the fission converter mechanical shutter, the reactor will be scrammed automatically upon loss of electric power to that shutter.

d) Shutters that control beam delivery and that are normally pneumatically-operated shall, in addition, be designed for manual closure.

e) It shall be possible to close the shutters that control beam delivery from within the medical therapy facility.

f) The fission converter mechanical shutter, which is normally operated electrically, shall also allow manual closure.

6. Each of the shutters that controls beam delivery shall be equipped with a light that indicates the status of the shutter. These lights shall be visible at each medical therapy facility's local control panel. In the event of a status light malfunction, it shall be acceptable to use the affected shutter provided that an alternate means of verifying position is available. Use of this alternate means of shutter position verification is limited to seven consecutive working days.

7. Each medical therapy facility shall be equipped with a monitor that provides a visual indication of the radiation level within the facility, that indicates both within the facility and at the local control panel, and that provides an audible alarm both within the facility and at the local control panel.

a) This radiation monitor shall be equipped with a backup power supply such as the reactor emergency power system or a battery.

b) This radiation monitor shall be checked for proper operation by means of a check source on the calendar day of and prior to any patient irradiation.

c) This radiation monitor shall be calibrated quarterly.

d) The audible alarm shall be set at or below 50 mrem/h. This monitor and/or its alarm may be disabled once the medical therapy room has been searched and secured, such as is done immediately prior to initiation of patient therapy. If this is done, the monitor and/or its alarm shall be interlocked so that they become functional upon opening of the medical therapy facility's shield door.

e) In the event that this monitor is inoperable, personnel entering the medical therapy facility shall use either portable survey instruments or audible alarm personal dosimeters as a temporary means of satisfying this provision. These instruments/dosimeters shall be in calibration as defined by the MIT Research Reactor's radiation protection program and shall be source-checked daily prior to use on any day that they are used to satisfy this provision. Use of these instruments/dosimeters as a temporary means of satisfying this provision is limited to seven consecutive working days.

8. An intercom or other means of two-way communication shall be operable both between each medical therapy facility control panel and the reactor control room, and also between each medical therapy facility control panel and the interior of the facility. The latter is for the monitoring of patients.

9. It shall be possible for personnel monitoring a patient to open each medical therapy facility's shield door manually.

10. It shall be possible to observe the patient through both a viewing port and by means of a closed-circuit TV camera. Both methods of patient visualization shall be operable at the outset of any patient irradiation. Should either fail during the irradiation, the treatment may be continued at the discretion of the BNCT physician authorized user. Adequate lighting to permit such viewing shall be ensured by the provision of emergency lighting.

11. The total radiation fluence delivered by the medical therapy facility beam as measured by on-line beam monitors shall not exceed that prescribed in the

patient treatment plan by more than 20%. The treatment is normally delivered in fractions in accordance with standard practice for human therapy. The 20% criterion applies to the sum of the radiation fluences associated with all fractions in a given treatment plan. A criterion of 30% applies to the difference between the administered and prescribed fluence for any given week (seven consecutive days). Finally, if the treatment consists of three or fewer fractions, then a criterion of 10% shall apply.

12. The following interlocks or channels shall be tested at least monthly and prior to treatment of human patients if the interlock or channel has been repaired or deenergized:

	Interlock or Channel	**Surveillance**
a)	Medical therapy facility minor scram	Channel Test
b)	Shutters will not open unless shield door is closed	Channel Test
c)	Shutters close upon both manual and automatic opening of shield door	Channel Test
d)	Shutters close and/or alarm on loss of electrical power and reduction of pressure in pneumatic operators, if applicable	Channel Test
e)	Manual closure of pneumatic shutters	Channel Test
f)	Shutters can be closed manually from within the facility	Channel Test
g)	Shutter status lights	Channel Test
h)	Radiation monitor alarm	Channel Test
i)	Radiation monitor and/or alarm enabled upon opening of shield door	Channel Test
j)	Intercoms	Channel Test
k)	Manual closure of fission converter mechanical shutter	Channel Test
l)	Availability of emergency power for beam monitor systems	Channel Test

In addition to the above, each medical therapy facility minor scram shall be tested prior to reactor startup if the reactor has been shut down for more than 24 hours.

13. Manual operation of each medical therapy facility's shield door in which the door is opened fully shall be verified semi-annually.

14. Use of BNCT Facility Beams

a) Use of the basement medical therapy facility beams shall be subject to the following:

(i) A functional check of the beam monitors that are described in provision 11 of this specification shall be made weekly for any week that the beam will be used for human therapy. This check shall be made prior to any patient irradiation for a given week. In addition, a functional check shall be performed prior to any patient irradiation in the event of a component replacement or a design modification.

(ii) A calibration check of the beam shall be performed every six months for any six-month interval that the beam will be used for human therapy. This six-month calibration check shall be made prior to any patient irradiation for a given six-month interval. In addition, a calibration check shall be performed prior to any patient irradiation in the event of a component replacement or a design modification.

(iii) A characterization of the beam shall be performed every twelve months for any twelve-month interval that the beam will be used for human therapy. This twelve-month characterization shall be made prior to any patient irradiation for a given twelve-month interval. A characterization shall also be performed prior to any patient irradiation in the event of a design modification. As part of the characterization process, the proper response of the beam monitors that are described in provision 11 of this specification shall be established.

(iv) The instruments (e.g., tissue-equivalent chamber and either a graphite or a magnesium wall ionization chamber or the equivalent) that are to be used to perform both calibration checks and characterization of the beam shall be calibrated by a secondary calibration laboratory. This calibration shall be performed at least once every two years for any two-year interval that the beam will be used for human therapy. The two-year calibration shall be made prior to any patient irradiation during any given two-year interval. (Note: If a method (e.g., foil activation) other than these checks is used for the calibration and or the characterization, then the devices (e.g., foils) used in that method shall either be traceable to the National Institute of Standards and Technology or be selected in accordance with the relevant ANSI/ANS standards.)

(v) There shall be a minimum of two neutron-sensitive beam monitors to initiate a patient irradiation. Once initiated, a patient irradiation may be continued at the discretion of both the certified medical physicist and the BNCT Principal Investigator, or designate, provided that at least one neutron-sensitive beam monitor is operable.

(vi) A calibration of the beam monitors that are described in provision 11 of this specification shall be performed at least once every two years for any two-year interval that the beam will be used for human therapy. The two-year calibration shall be made prior to any patient irradiation during any given two-year interval.

b) Use of the fission converter medical therapy facility beam shall be subject to the following:

(i) Functional checks: the same requirements as provision 14(a)(i) above.

(ii) Calibration checks: the same requirements as provision 14(a)(ii) above except that all frequencies are weekly instead of six months.

(iii) Characterization: the same requirements as provision 14(a)(iii) above except that all frequencies are six months instead of twelve months.

(iv) Instrument calibration: the same requirements as provision (14(a)(iv).

(v) Number of beam monitors: the same requirements as provision 14(a)(v).

(vi) Calibration of beam monitors: the same requirements as provision 14(a)(vi).

15. Maintenance, repair, and modification of the medical therapy facilities shall be performed under the supervision of a senior reactor operator who is licensed by the U.S. Nuclear Regulatory Commission to operate the MIT Research

Reactor. The 'medical therapy facility' includes the beam, beam shutters, beam monitoring equipment, medical therapy facility shielding, shield door, and patient viewing equipment. All modifications will be reviewed pursuant to the requirements of 10 CFR 50.59. The operating couch, patient positioning equipment, medical instruments, and other equipment used for the direct medical support of the patient are not considered part of the medical therapy facility for purposes of this provision, except insofar as radiation safety (i.e., activation and/or contamination) is concerned.

16. Personnel who are not licensed to operate the MIT Research Reactor but who are responsible for either the medical therapy or the beam's design including construction and/or modification may operate the controls for the corresponding medical therapy facility beam provided that:

    a) Training has been provided and proficiency satisfactorily demonstrated on the design of the facility, its controls, and the use of those controls. Proficiency shall be demonstrated annually.

    b) Instructions are posted at the medical therapy facility's local control panel that specify the procedure to be followed:

        (i) to ensure that only the patient is in the treatment room before turning the primary beam of radiation on to begin a treatment;

        (ii) if the operator is unable to turn the primary beam of radiation off with controls outside the medical therapy facility, or if any other abnormal condition occurs. A directive shall be included with these instructions to notify the reactor console operator in the event of any abnormality.

    c) In the event that a shutter affects reactivity (e.g., the D_2O shutter for the medical room below the reactor and the converter control shutter for the fission converter beam), personnel who are not licensed on the MIT Research Reactor but who have been trained under this provision may operate that shutter provided that verbal permission is requested and received from the reactor console operator immediately prior to such action. Emergency closures are an exception and may be made without first requesting permission.

Records of the training provided under subparagraph (a) above shall be retained in accordance with the MIT Research Reactor's training program or at least for three years. A list of personnel so qualified shall be maintained in the reactor control room.

17. Events defined as 'recordable' under definition 8 of this specification shall be recorded and the record maintained for five years. Events defined as 'misadministrations' under definition 9 of this specification shall be reported to the U.S. Nuclear Regulatory Commission (24 hours verbal, 15 day written report). The 24 hour verbal reports will be made to the Headquarters Operation Center. The 15 day written reports will be sent to the NRC Document Control Desk.

18. The requirements of the Quality Management Program (QMP) for the Generation of Medical Therapy Facility Beams for Human Therapy at the Massachusetts Institute of Technology Research Reactor shall be observed for any human therapy. (Note: The presence of this commitment to observe the QMP in these specifications does not preclude modifying the QMP as provided in that document. Any such modifications are not considered to be a change to the MITR-II Technical Specifications.)

19. Reactor facilities (e.g., prompt gamma for the determination of boron concentration in blood or tissue) that are used to perform measurements associated with the conduct of medical therapy shall be calibrated every twelve months for any twelve-month interval that the beam will be used for human therapy. This twelve-month calibration shall be made prior to any patient irradiation for a given twelve-month interval. This calibration could be

done by measuring a series of standards that span the anticipated range of boron in blood or tissue. In addition, a single point check, (e.g., verification that a single standard is measured ±10% of its true value) shall be performed prior to any patient irradiation.

20. An emergency power source shall be available for the beam monitor systems.

<u>Definitions</u>

1. The medical therapy facilities are equipped with shutters that are used (i) to control beam delivery and (ii) to adjust the neutron energy spectrum of the beam. The former currently include lead, boral, and light water shutters as described in Reference 6.5-1. The heavy water blister tank, which is also described in Reference 6.5-1, is an example of the latter. It is conceivable that these designations may change should it be found desirable to alter the beam configuration. Accordingly, the phrase "shutters that control beam delivery" refers either to the aforementioned existing shutters or to any future shutter or group thereof that provides an equivalent or greater reduction in beam intensity. Shutter-effect analyses shall be documented through the standard safety review process including, where appropriate, an SAR revision and submission to NRC under 10 CFR 50.59.

2. The term 'calibration check' refers to the process of checking the beam intensity and quality via one or more of the following: foil activation; use of a fission chamber; use of an ion chamber; or an equivalent process. The purpose of a calibration check is to ensure that the beam has not changed in a significant way (e.g., energy spectrum or intensity) from the beam that was characterized.

3. The term 'functional check of the beam monitors' shall consist of verifying that system output is consistent (± 10%) with previously measured values upon normalization to a common neutronic power level.

4. The term 'characterization' refers to the process of obtaining the dose-versus-depth profile in phantoms as described in Reference 6.5-2 or an equivalent process. The dose-versus-depth profile from the surface of the phantom to a depth at least equivalent to the total thickness of the body part to be treated on a central axis is deemed adequate for a characterization. Fast neutron, thermal neutron, and gamma ray components are determined in a characterization and monitors are normalized by this characterization.

5. The term ‘component replacement’ means the replacement of a component in the beam with an identical unit or the re-installation of a component in the beam for which a characterization has already been performed. For example, the latter may be a change of collimators.

6. The term ‘design modification’ as applied to a medical therapy facility beam refers (a) to a change that is shown to alter the dose-versus-depth profile of the fast neutrons, thermal neutrons, or gamma rays in the beam as sensed by the calibration check and (b) to a change that has the potential to increase significantly the amount of activation products in the medical therapy facility when the beam is to be used for the treatment of human patients.

7. The term ‘radiation fluence’ means the total fluence of neutrons and gamma radiation that is emitted in a medical therapy facility beam. The determination of the ratios of gamma, fast neutron, and thermal neutron fluences is part of

the beam characterization. Knowledge of these ratios allows the total radiation fluence to be monitored by the on-line detectors, which are neutron-sensitive. Compliance with the limits specified on radiation fluence by this specification is determined by reference to the fluence monitored by these detectors.

8. The term ‘recordable event’ means the administration of:

   a) A radiation treatment without a written directive; or

   b) A radiation treatment where a written directive is required without reporting to the medical use licensee in writing each fluence given within 24 hours of the treatment; or

   c) A treatment delivery for which the administered radiation fluence for any given fraction is 15% greater than prescribed.

9. The term ‘misadministration’ means the administration of a radiation therapy:

   a) Involving the wrong patient, wrong beam (basement or fission converter), or wrong treatment site; or

   b) When the treatment delivery is not in accordance with provision 11 of this specification.

10. The term 'written directive' means an order in writing for a specific patient, dated and signed by a BNCT physician authorized user prior to the administration of radiation and which specifies the treatment site, the total radiation fluence, radiation fluence per fraction, the medical facility (basement medical therapy facility beam or fission converter medical therapy facility beam) and collimator, if any, to be used, and overall treatment period.

11. The term 'human therapy' means radiation treatments that are of direct therapeutic benefit to the patient and/or part of investigatory studies that involve humans.

12. The term 'BNCT physician authorized user' means a medical physician authorized by the medical use licensee's radiation safety committee to act as an authorized user for BNCT on humans.

13. The term 'certified medical physicist' means a medical physicist certified in either radiological physics or therapeutic radiation physics by the American Board of Radiology, or in therapeutic radiation physics by the American Board of Medical Physics and who also has specific training in neutron dosimetry and neutron capture therapy.

14. The term “BNCT Principal Investigator” means a person representing MIT who holds an advanced degree in science or engineering and who has two or more years of experience in BNCT.

15. The term “basement medical therapy facility beam” means the beam emanating from the MIT Research Reactor into the medical therapy room that is physically located below the reactor on the building’s lower level.

16. The term “fission converter medical therapy facility beam” means the beam emanating from the MIT Research Reactor’s fission converter into the medical therapy facility that is physically located adjacent to the reactor on the building’s main floor.

17. The term “calibration of the beam monitors” refers to the process whereby the beam monitors that are described in provision 11 of this specification are calibrated against instruments that measure dose including a tissue-equivalent chamber and either a graphite- or magnesium-wall ionization chamber (or the equivalent to any of these three) that have in turn been calibrated by a secondary calibration laboratory.

Basis

The stipulation that patients only be accepted from a medical use licensee that has an NRC or an Agreement State medical use license that contains BNCT specific conditions and commitments for BNCT treatment of humans conducted at the Massachusetts Institute of Technology Research Reactor's Medical Therapy Facilities ensures that medical criteria imposed by NRC or the Agreement State on such licensees for the use of the MIT Research Reactor's medical therapy facility beams for human therapy will be fulfilled. The second provision delineates the division of responsibilities between the Massachusetts Institute of Technology and the medical licensee that refers the patient. Also, it establishes administrative authority and protocol for initiating and terminating a radiation therapy.

The requirement that it be possible to initiate a minor scram from control panels located in the medical therapy facility areas ensures the attending physician and/or medical physicist of the capability to terminate the treatment immediately should the need arise. The provision that access to each medical therapy facility be limited to a single door ensures that there will be no inadvertent entries. The various interlocks for the shutters that control beam delivery ensure that exposure levels in the medical therapy facility will be minimal prior to entry by personnel who are attending the patient. The shutter-indication lights serve to notify personnel of the beam's status. The provision for a radiation monitor ensures that personnel will have information available on radiation levels in the medical therapy facility prior to entry. The purpose of this monitor's audible alarm is to alert personnel to the presence of elevated radiation levels, such as exist when the shutters that control beam delivery are open. This monitor and/or its alarm may be disabled once the medical therapy facility has been searched and secured so that it will (1) not disturb a patient and (2) not distract attending personnel. The monitor and/or its alarm are interlocked with the shield door so that they are made functional upon opening that door, and hence prior to any possible entry to the medical therapy facility. One intercom provides a means for the prompt exchange of information between medical personnel and the reactor operator(s). The second intercom is for monitoring the patient.

The provision for manual operation of each medical therapy facility's shield door ensures access to any patient in the event of a loss of electrical power. The presence of a viewing window and a closed-circuit TV camera provide the attending BNCT physician authorized user and/or medical physicist with the opportunity to monitor the patient visually as well as through the use of various instruments. The viewing window will function even during an electric power failure because of the provision for emergency lighting.

The specification that the total radiation fluence for a therapy (i.e., the radiation fluences for the sum of all fractions specified in a given treatment plan) not exceed that prescribed in the patient treatment plan by 20% establishes a trigger limit on the delivered fluence above which NRC has to be notified of a misadministration. The 20% criterion is based on the definition of misadministration (clause 4(iv)) as given in 10 CFR 35.2. The criterion that the difference between the administered and prescribed fluence for any seven consecutive days is set at 30%. This is also in accordance with the definition of misadministration (clause 4(iii)) as given in 10 CFR 35.2. Finally, if a treatment involves three or fewer fractions, then a more stringent criterion, 10%, applies to the difference between the total radiation fluence for a therapy and that prescribed in the treatment plan (10 CFR 35.2(4ii)). The surveillance requirements for the functional checks as well as those for the beam calibration checks and characterizations provide a mechanism for ensuring that each medical therapy facility and its beam will perform as originally designed. Similarly, the surveillance requirements on the instruments used to perform these checks and characterizations ensure that these instruments are calibrated by a means traceable to the National Institute of Standards and Technology. The chambers specified (tissue-equivalent, and graphite or magnesium-wall) were chosen because they measure dose as opposed to fluence. Finally, the requirement on the number of beam monitors is in keeping with standard practice for gamma-ray sources.

The specification on maintenance and repair of the medical therapy facilities ensures that all such activities are performed under the supervision of personnel cognizant of quality assurance and other requirements such as radiation safety. The provision on the training and proficiency of non-licensed personnel ensures that all such personnel will receive instruction equivalent to that given to licensed reactor operators as regards use of the medical therapy facility beams. (Note: Licensed reactor operators may, of course, operate the medical therapy facility beams.) Also, this provision provides for the posting of instructions to be followed in the event of an abnormality.

The specification on 'recordable events' and 'misadministrations' provides for the documentation and reporting to the U.S. Nuclear Regulatory Commission of improper events regarding the generation and use of medical therapy facility beams. The requirement that the Quality Management Program (QMP) be observed ensures that radiation treatments provided by a medical therapy facility beam will be administered as directed by the BNCT physician authorized user.

The specification on calibration of reactor facilities that are used to measure the concentration of boron in blood or tissue ensures that these measurements are accurate.

References

6.5-1 MITR Staff, "Safety Analysis Report for the MIT Research Reactor (MITR-II)," Report No. MITNE-115, 22 Oct. 1970, Section 10.1.3.

6.5-2 Choi, R.J., "Development and Characterization of an Epithermal Beam for Boron Neutron Capture Therapy at the MITR-II Research Reactor," Ph.D. Thesis, Nuclear Engineering Department, Massachusetts Institute of Technology, April 1991.

Quality Management Program

for

Generation of MITR-II Medical Therapy Facility Beams

for Human Therapy

Quality Management Program

1. Purpose: The objective of this quality management program is to ensure that radiation treatments provided by the MIT Research Reactor's (MITR-II) Medical Therapy Facility beams will be administered as directed by a BNCT physician authorized user.

2. Authorized Medical Use Licensees: Use of the MIT Research Reactor's Medical Therapy Facility beams, for the treatment of human subjects, is limited to the BNCT physician authorized users from medical centers with an NRC or Agreement State medical use license that contains BNCT specific conditions and commitments for BNCT treatment on humans conducted at the Massachusetts Institute of Technology Research Reactor's Medical Therapy Facilities.

3. Program Requirements: The following requirements are established as part of this quality management program:

    a) A written directive will, except as noted in subparagraph (iv) below, be prepared by a BNCT physician authorized user of either the NRC or Agreement State medical use licensee prior to the administration of any radiation therapy. This directive shall be written, signed, and dated by the BNCT physician authorized user and it shall include the following information:

        (i) Name and other means of identifying the patient.

        (ii) Name of the BNCT physician authorized user and certified medical physicist in charge of the therapy.

        (iii) The total radiation fluence to be administered, the radiation fluence per fraction, the treatment site, and the overall treatment period.

        (iv) If, because of the patient's condition, a delay in order to provide a written revision to an existing written directive would jeopardize the patient's health, an oral revision to an existing written directive will be acceptable, provided the oral revision is documented immediately in the patient's record and a revised written directive is signed by a BNCT physician authorized user within 48 hours of the oral revision.

        Also, a written revision to an existing written directive may be made for any therapeutic procedure provided that the revision is dated and signed by a BNCT physician authorized user prior to the administration of the next fraction.

        If, because of the emergency nature of the patient's condition, a delay in order to provide a written directive would jeopardize the patient's health, an oral directive will be acceptable, provided that the information contained in the oral directive is documented immediately in the patient's record and a written directive is prepared within 24 hours of the oral directive.

(v) In order to ensure that the Staff of the MIT Research Reactor has the most recent written directive from the medical use licensee and the correct directive for the patient in question, a copy of that directive shall be hand-delivered to the MITR-II Staff by the Staff of the medical use licensee who accompany the patient to MIT. This copy shall then be checked against the most recent previous transmission. Any discrepancy shall be resolved by the medical use licensee prior to the initiation of patient irradiation.

(vi) The BNCT Principal Investigator, or designate, will date and sign the written directive to verify that current and accurate beam characteristic parameters were provided to the NRC or Agreement State medical use licensee as appropriate and that the radiation fluence desired in the written directive was delivered. A copy of this signed directive shall be provided to the medical use licensee within twenty-four hours of a treatment.

b) Prior to each administration of any radiation, the patient's identity will be verified by more than one method as the individual named in the written directive. The MIT Nuclear Reactor Laboratory will use any two or more of the following acceptable methods of identification:

(i) Self-identification by patients who are conscious upon arrival at the MIT Research Reactor. Information provided by the patient shall include any two of the following: name, address, date of birth, or social security number. The information provided by the patient is to be compared to the corresponding information in the patient's record.

(ii) Hospital wrist band identification with the wrist band information to be compared to the corresponding information in the patient's record.

(iii) Visual identification against photographs provided with the written directive.

(iv) Other methods as specified in U.S. Nuclear Regulatory Commission Regulatory Guide 8.33, "Quality Management Program."

c) The plan of treatment is certified by the certified medical physicist to be in accordance with the written directive. In this regard, the Massachusetts Institute of Technology is responsible for calibrating the output of the beam monitoring instrumentation versus dose in phantom and for providing a central axis dose versus depth profile. This information will then be used by personnel at either the NRC or the Agreement State medical use licensee as appropriate to generate a plan of treatment. Conformance of the beam to its design characteristics is confirmed through the measurements specified in MITR-II Technical Specification #6.5, "Generation of Medical Therapy Facility Beams for Human Therapy." Functional checks are made of the beam monitors at least weekly for any week that the beam will be used for human therapy (provision 14(a)). Calibration checks are performed every six months for any six-month interval that the beam will be used for human therapy (provision 14(b)). Each beam is characterized dosimetrically every twelve months (provision 14(c)). The instruments that are used to perform calibration checks and characterization of the beams are calibrated every two years by a secondary calibration laboratory (provision 14(d)).

d) Each administration of radiation is in accordance with the written directive subject to the tolerances established in provision 11 of MITR-II Technical Specification #6.5, "Generation of Medical Therapy Beams for Human Therapy."

e) Any unintended deviations from the written directive shall be identified and evaluated, and appropriate action taken. Such action shall include informing the medical use licensee of the deviation. These reviews shall be performed monthly for any month in which human therapy was conducted. For each patient case reviewed, it shall be determined whether the administered total fluence, fluence per fraction, treatment site, and overall treatment period were as specified in the written directive. In the event of any deviation from the written directive, the licensee (MIT) shall identify its cause and the action required to prevent recurrence. These actions may include new or revised policies, new or revised procedures, additional training, increased supervisory review of work, or other measures as deemed appropriate. Corrective actions shall be implemented as soon as practicable.

4. Program Implementation: The following practices shall be observed in order to ensure proper implementation of the quality management program:

a) A review shall be conducted of the quality management program. This review shall include, since the last review, an evaluation of:

(i) A representative sample of patient administrations,

(ii) All recordable events, and

(iii) All misadministrations.

The objective of this review is to verify compliance with all aspects of the quality management program. For purposes of this review, the term 'representative' in statement (i) above is defined as 100% sampling up to twenty patients; a sample of twenty for twenty-one to one hundred patients, and 20% sampling for more than one hundred patients. In order to eliminate any bias in the sample, the patient cases to be reviewed should be selected randomly.

b) The procedure for conducting the above review is as follows:

(i) The review shall be performed by the Director of the MIT Radiation Protection Program or his designate.

(ii) The review shall be performed annually.

(iii) Patient administrations selected for review shall be audited to determine compliance with each of the requirements listed in paragraph (3) above.

(iv) The review shall be written and any items that require further action shall be so designated. Copies of the review shall be provided to the NRL Managing Director for Operations and to the MIT Reactor Safeguards Committee who will evaluate each review and, if required, recommend modifications in this quality management program to meet the requirements of paragraph (3) above. A copy of these reviews will also be provided to each medical use licensee.

c) Records of each review, including the evaluations and findings of the review, shall be retained in an auditable form for three years.

d) The licensee (MIT) shall reevaluate the Quality Management Program's policies and procedures after each annual review to determine whether the program is still effective or to identify actions required to make the program more effective.

5. Response to Recordable Event: Within thirty days after the discovery of a recordable event, the event shall be evaluated and a response made that includes:

a) Assembling the relevant facts, including the cause;

b) Identifying what, if any, corrective action is required to prevent recurrence; and

c) Retaining a record, in an auditable form, for three years, of the relevant facts and what corrective action, if any, was taken.

A copy of any recordable event shall be provided to the affected medical use licensee.

6. Records Retention: The following records shall be retained:

a) Each written directive for three years; and

b) A record of each administered radiation therapy where a written directive is required in paragraph (3(a)) above, in an auditable form, for three years after the date of administration.

7. Program Modification: Modifications may be made to this quality management program to increase the program's efficiency provided that the program's effectiveness is not decreased. All medical use licensees shall be notified of any modifications and provided with a copy of the revised program. The licensee (MIT) shall furnish the modification to the NRC (Region I) within 30 days after the modification has been made.

8. Report and Surveillance Frequency: Any report or other function that is required to be performed in this Quality Management Program at a specified frequency shall be performed within the specified time interval with:

a) a maximum allowable extension not to exceed 25% of the specified surveillance interval, unless otherwise stated in this Quality Management Program;

b) a total maximum combined interval time for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

9. Definitions:

a) The term 'BNCT physician authorized user' means a medical physician authorized by the medical use licensee's radiation safety committee to act as an authorized user for BNCT on humans.

b) The term 'certified medical physicist' means a medical physicist certified in either radiological physics or therapeutic radiation physics by the American Board of Radiology, or in therapeutic radiation physics by the American Board of Medical Physics and who also has specific training in neutron dosimetry and neutron beam capture therapy.

c) The term "BNCT Principal Investigator" means a person representing MIT who holds an advanced degree in science or engineering and who has two or more years of experience in BNCT.

10. Applicability: This Quality Management Program applies solely to the generation of medical therapy facility beams for the treatment of human subjects. It does not apply to any other use of the medical therapy facilities and/or their beams. Reports and surveillances listed in this specification are only required if human therapy was conducted during the referenced interval.

6.6 Design and Operation of the Fission Converter Facility

Applicability

This specification applies to the operation of the Fission Converter Facility. It does not pertain to the use made of the fission converter beam nor does it apply to the associated medical therapy facility. Use of that facility for the treatment of human patients and/or investigatory studies that involve humans shall be in accordance with the provisions of Specification 6.5 and its associated quality management program.

The provisions of this specification are only applicable if fuel is present in the fission converter tank.

Organization

This specification contains five subsections. These are:

6.6.1 Safety Limits and Limiting Safety System Settings

6.6.2 Limiting Conditions for Fission Converter Operation

6.6.3 Fission Converter Surveillance Requirements

6.6.4 Fission Converter Design Features

6.6.5 Reporting Requirements

Definitions

1. Fission Converter Shutdown

That condition where the converter control shutter is fully inserted or where the reactor is in a shutdown condition. The fission converter is considered to be operating whenever this condition is not met.

2. Fission Converter Secured

The overall condition where there is no fuel in the fission converter or where all of the following conditions are satisfied:

a) The fission converter is shut down,

b) The converter control shutter (CCS) control panel key switch is in the off position and the key is in proper custody, and

c) There is no work in progress within the converter tank involving fuel.

6.6.1 Safety Limits and Limiting Safety System Settings

6.6.1.1 Safety Limits

Applicability

This specification applies to the interrelated variables associated with fission converter thermal and hydraulic performance. These variables are the fission converter neutronic power (P), the steady-state average primary coolant outlet temperature (T_{out}) if under forced convection, the fission converter tank coolant mixing temperature (T_{mix}) if under natural circulation, the fission converter primary coolant flow rate (W_p), and the fission converter primary coolant height above top of the fuel elements in the main tank (H). For forced convection, the fission converter shall contain either ten or eleven fuel elements. For natural convection, the fission converter shall contain eleven fuel elements.

Objective

To establish limits within which the integrity of the fuel clad is maintained.

Specification

1. For forced convection, the point determined by the true values of P, W_p, and T_{out} shall not be above the line given in Figure 6.6.1.1-1 corresponding to the coolant height, H.
2. For natural convection, the coolant height shall be at 2.4 m or higher and the point determined by the true values of P and T_{mix} shall not be above the line given in Figure 6.6.1.1-2.

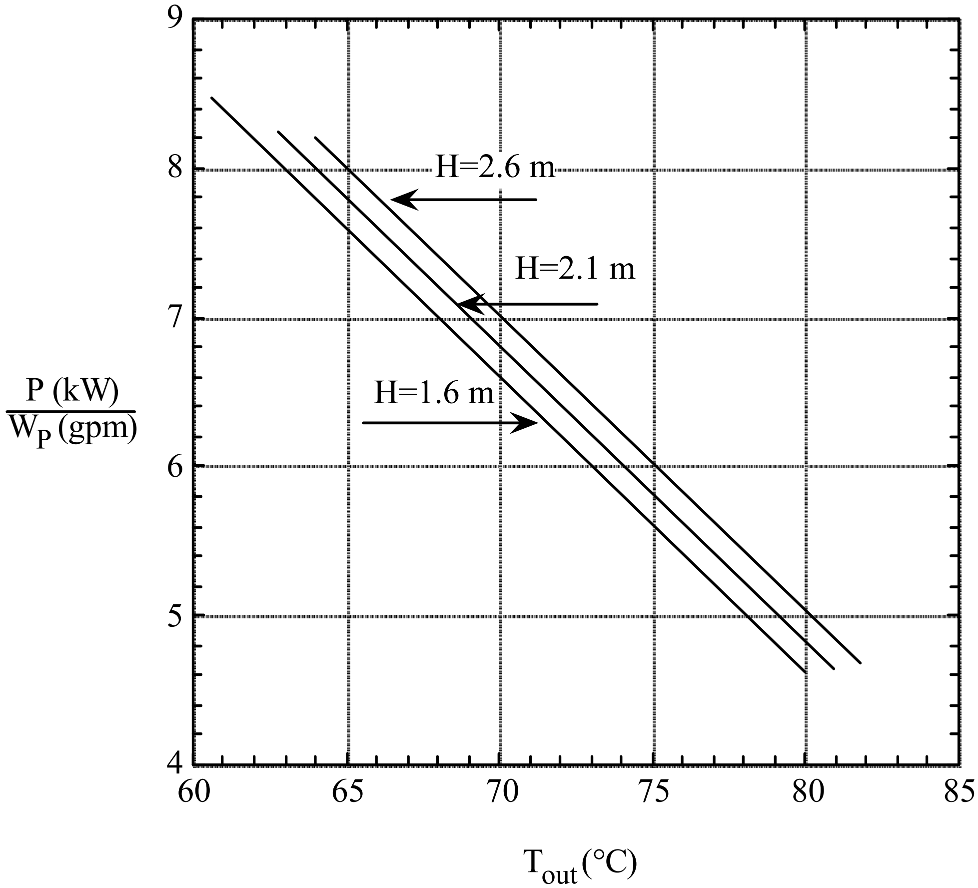

Figure 6.6.1.1-1 Fission Converter Safety Limits for Forced Convection.
(for either ten or eleven fuel elements)

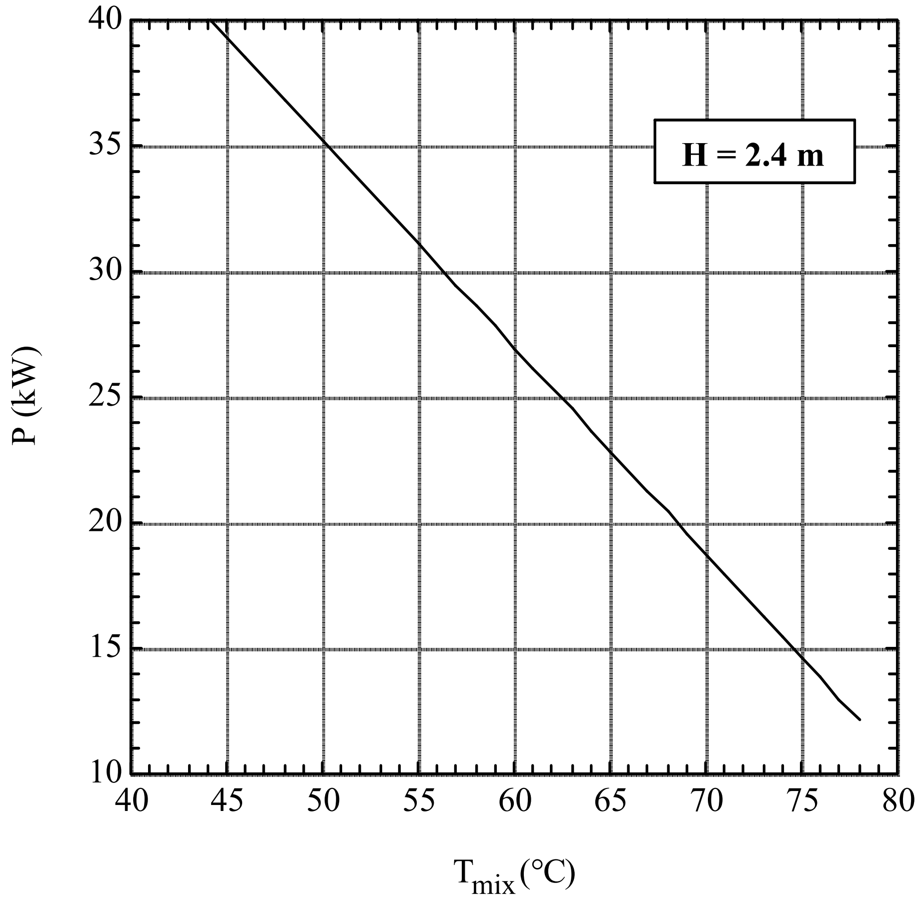

Figure 6.6.1.1-2 Fission Converter Safety Limits for Natural Convection. (for eleven fuel elements only)

Basis

In the MITR-II SAR it is noted that critical heat flux is a conservative limit beyond which fuel damage may occur from overheating. In addition, the onset of multichannel flow instability (OFI) can lower the burnout heat flux. However, OFI is a complicated phenomenon and the effects of heat flux spatial distributions are not taken into account in the correlations developed for OFI. Onset of significant voiding (OSV), on the other hand, can be more accurately predicted for various heat flux spatial distributions. OSV describes the condition where the bubbles grow larger on a heated surface and detach regularly to the main flow stream. It has been observed experimentally that OSV occurs before OFI. Therefore, OSV is conservatively assumed as the criterion for the safety limits of the fission converter.

OSV was calculated in the Fission Converter Safety Evaluation Report (SER) for the hot channel. Uncertainties because of departure from nominal design specifications, measurement errors, and use of empirical correlations are taken into account in these calculations. The safety limits were evaluated based on the limiting core operating conditions described in Specification 6.6.2.1. Figure 6.6.1.1-1 shows the calculated fission converter safety limits for forced convection for three coolant heights: 2.6 m, 2.1 m, and 1.6 m above the top of the fuel elements. This figure is applicable to either ten or eleven fuel elements.

The purpose of allowing fission converter operation at low power in the absence of forced convection is to facilitate activities such as flux measurements in the fueled region. Natural circulation is achieved by removing the inlet pipes, which are used for forced convection, from the downcomers. Calculations show that the natural circulation is sufficient to dissipate the energy that is generated provided that the limit on the fission converter tank coolant mixing temperature is not exceeded. OSV for natural circulation was calculated for a coolant height of 2.4 m. This coolant height corresponds to the top of the downcomers. The result for eleven elements is shown in Figure 6.6.1.1-2.

6.6.1.2 Limiting Safety System Settings (LSSS)

Applicability

This specification applies to the setpoints for the safety channels monitoring the fission converter neutronic power (P), the steady-state average primary coolant outlet temperature (T_{out}) if under forced convection, the fission converter tank coolant mixing temperature (T_{mix}) if under natural circulation, the fission converter primary coolant flow rate (W_p), and the fission converter primary coolant height above top of fuel elements in the main tank (H). For forced convection, the fission converter shall contain either ten or eleven fuel elements. For natural convection, the fission converter shall contain eleven fuel elements.

Objective

To ensure that automatic protective actions will prevent the onset of nucleate boiling in the fission converter fueled region and will thus prevent operating conditions from exceeding the safety limit.

Specification

1. The measured values of the limiting safety system settings on P, W_p, T_{out}, and H for fission converter operation with forced convection shall be as follows:

Variable	**Limiting Safety System Setting**
P	300 kW (max)
W_p	45 gpm (min)
T_{out}	60 °C (max)
H	2.1 m above top of fuel elements (min)

2. The fission converter may be operated at power levels up to 20 kW in the absence of forced convection, provided that the inlet pipes are removed so as to allow natural circulation. The measured values of the limiting safety system settings on P, T_{mix}, and H for fission converter operation with natural circulation shall then be as follows:

Variable	**Limiting Safety System Setting**
P	20 kW (max)
T_{mix}	60 °C (max)
H	2.5 m above top of fuel elements (min)

Basis

The limiting safety system settings (LSSS) are established to allow a sufficient margin between normal operating conditions and the safety limits, so that automatic shut down actions will ensure that the fission converter is maintained in a safe condition during normal operation. Onset of nucleate boiling (ONB) is chosen as the criterion for the LSSS derivation. ONB (also called incipient boiling) defines the condition where bubbles first start to form on the heated surface. Because most of the liquid is still subcooled, the bubbles do not detach but grow and collapse while attached to the wall. LSSS are chosen so that boiling will not occur anywhere in the fueled region as long as the limits are not exceeded.

The ONB is calculated in the Fission Converter SER for the hot channel. Uncertainties because of departure from nominal design specification, measurement errors, and the use of empirical correlations are taken into account in these calculations. The LSSS were evaluated based on the limiting core operating conditions described in Specification 6.6.2.1.

Figure 6.6.1.2-1 shows the result of the fission converter LSSS calculations for a primary coolant flow rate of 45 gpm and a coolant height of 2.1 m for operation with forced convection. The LSSS temperature calculated for 300 kW is 63°C, and hence a primary coolant outlet temperature setting of 60°C is conservative.

For fission converter operation with natural circulation, calculations have shown that the prediction of ONB coincides with that of OSV because of the low flow rate. Therefore, a 5°C margin is added to the safety limit curve to establish the LSSS. This 5°C margin corresponds to about 6 minutes of heat up time in the mixing area with the fission converter at 20 kW and thus provide adequate response time for corrective actions. The resulting LSSS curve is shown in Figure 6.6.1.2-2. The LSSS for fission converter operation with natural circulation is conservatively determined for a maximum power of 20 kW and a maximum coolant mixing temperature of 60°C with a coolant height of 2.5 m.

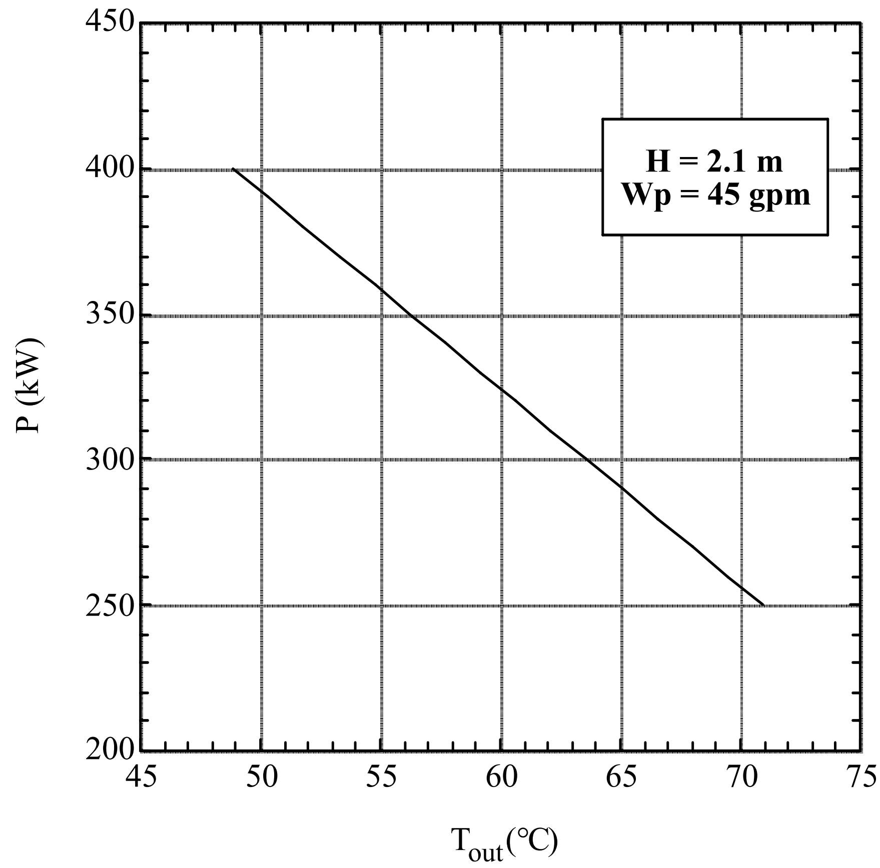

Figure 6.6.1.2-1 Calculated Results for the Fission Converter LSSS for Operation with Forced Convection. (for either ten or eleven fuel elements)

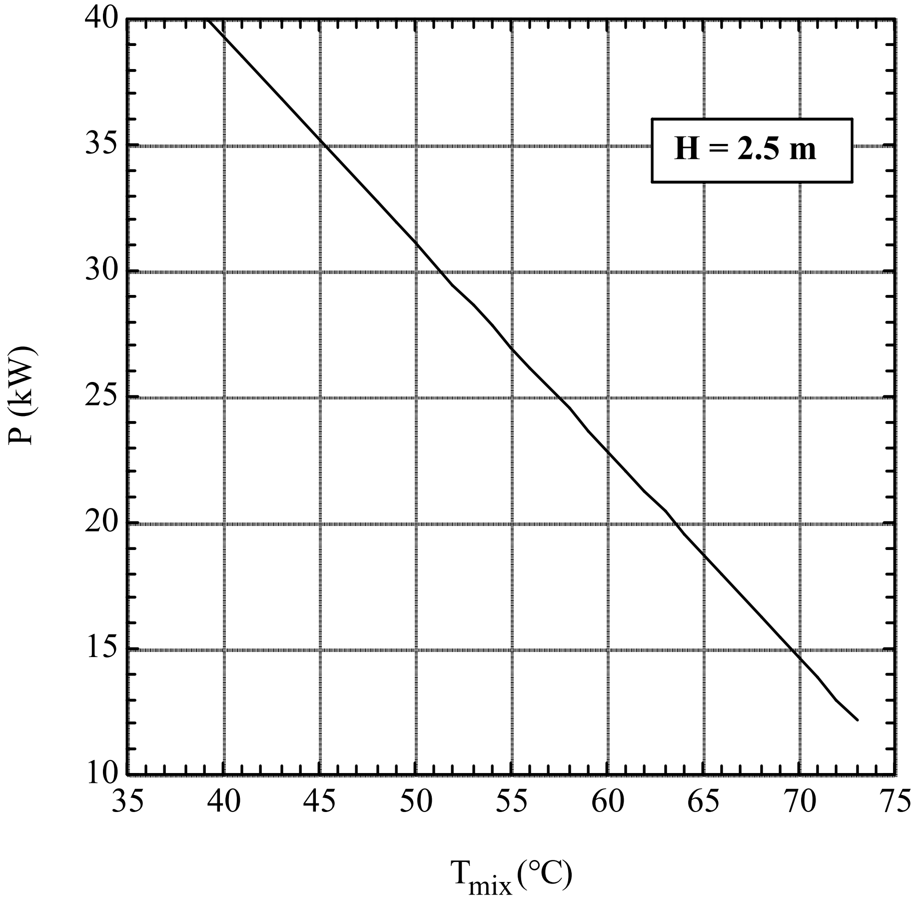

Figure 6.6.1.2-2 Fission Converter LSSS for Operation with Natural Convection (for eleven fuel elements only)

6.6.2 Limiting Conditions for Fission Converter Operation

6.6.2.1 Limiting Operating Conditions for the Fueled Region

Applicability

This specification applies to the fission converter core operating conditions. The variables used to define the core operating conditions are:

- F_p the fraction of the total power deposited in the fueled region (both fuel and coolant),
- F_{HC} the ratio of the maximum power deposited in the hottest fuel plate to the average power per fuel plate,
- F_f the ratio of the primary coolant flow that goes through the fueled region to the total primary coolant flow, and
- d_f the ratio of the minimum flow to the average flow in the coolant channels.

Objective

To ensure that the operating parameters are maintained within the bounds that are used to establish the safety limits and the limiting safety system settings of the fission converter.

Specification

1. $F_p \times F_{HC} \leq 1.53$
2. $F_f \times d_f \geq 0.80$
3. After each change in loading of the fueled region which might increase the hot channel factor, an evaluation shall be made to ensure that Specification 6.6.2.1.1 above is satisfied. A record of these evaluations shall be approved by two licensed SROs.
4. All positions in the fueled region shall be filled with either a fuel element or another approved unit. Specifications 6.6.2.1.1 and 6.6.2.1.2, the safety limits, and the LSSS shall be re-evaluated for:

a) forced convection with other than ten or eleven fuel elements in the fueled region, and

b) natural convection with other than eleven fuel elements in the fueled region.

5. The maximum fuel burnup shall be limited in accordance with Specification 3.1.6.3.

6. The maximum allowed value of k_{eff} for the fueled region shall not exceed 0.90.

7. Fuel elements contained in the fueled region of the fission converter shall be oriented so that the plates are "edge on" towards the MITR-II core.

8. The fission converter tank lid shall be in place and sealed for operation greater than 20 kW.

Basis

The fission power deposition factor (F_p), the hot channel factor (F_{HC}), the fueled region coolant flow distribution factor (F_f), and the channel flow disparity factor (d_f) are all dependent on the fission converter fueled region design. Specifications 6.6.2.1.1 and 6.6.2.1.2 that describe the limits on these factors are conservatively determined for the fission converter design and provide reasonable margin for deviations. These factors form the basis for the thermal-hydraulic limits calculations and they should be verified during initial startup of the fission converter.

Safety limits and limiting safety system settings of the fission converter described in Specification 6.6.1 are derived based on eleven fuel elements in the fueled region. Calculations showed that the same SL and LSSS curves (Figures 6.6.1.1-1 and 6.6.1.2-1) can be used for forced convection operation with ten fuel elements. However, there is a significant effect on the SL and LSSS curves for natural convection operation with other than eleven fuel elements.

Therefore, these limits shall be re-evaluated for natural convection operation if other than eleven fuel elements are to be used.

The maximum fuel burnup density is chosen in accordance with Specification 3.1.6.3. This limit was developed based on the MITR-II fuel design.

The effective multiplication factor (k_{eff}) for the fission converter was calculated in the SER for different combinations of coolant and fuel element U-235 content using the Monte Carlo N-Particle (MCNP) code. The k_{eff} values calculated for a D_2O cooled system are 0.268 for partially spent MITR-II fuel and 0.344 for fresh MITR-II fuel. For an H_2O cooled system, the k_{eff} calculated values are 0.514 and 0.618 for partially spent and fresh MITR-II fuel, respectively. Because the k_{eff} predicted is much smaller than unity, a criticality accident is not credible. The criterion of a k_{eff} less than 0.90 was chosen because it is in accordance with Specification 5.4 for fuel storage locations.

Calculations show that the power peaking in the hot channel would be unacceptable if the elements were to be rotated so that fuel plates were facing the MITR-II. Therefore, administrative procedures will be used to ensure that fuel elements are loaded with an "edge-on" orientation.

Operation of the fission converter without the top shield lid in place is allowed for power levels up to 20 kW. Calculations in the fission converter SER have shown that the estimated dose rate at that power level is 450 mR/h at the coolant surface with a coolant height of 2.4 m. This dose rate is not in excess of those occasionally encountered during certain maintenance operations, and it has been demonstrated that administrative actions can provide adequate controls under such conditions. Adequate controls will be instituted during such experiments to prevent excessive personnel exposure.

6.6.2.2 Maximum Allowed Reactivity Addition from the Converter Control Shutter

Applicability

This specification applies to the reactivity worth of the converter control shutter.

Objective

To ensure that the integrity of the MITR-II fuel is maintained during operation of the fission converter.

Specification

1. The reactivity worth of the converter control shutter shall be determined in the initial startup testing of the fission converter and verified annually thereafter.
2. The reactivity worth of the converter control shutter shall be in accordance with Specification 6.1.

Basis

MITR-II Technical Specifications provide several approaches for limiting the reactivity associated with an experimental facility. MITR-II Technical Specification 6.1 imposes limits depending on whether the experiment is classified as moveable, non-secured, or secured. Accordingly, this approach is used for the fission converter. The reactor's routine controls can be used to negate any reactivity insertion that results from opening of the converter control shutter, provided that the reactivity is less than the limit specified for a movable experiment.

6.6.2.3 Fission Converter Fuel Element Security, Storage, and Handling

Applicability

This specification applies to the security, storage, and handling of the fission converter fuel elements.

Objective

To ensure that the fueled elements will be properly stored and handled in a manner to protect the safety of reactor personnel.

Specification

1. All fuel elements used in the fission converter fueled region shall be maintained self-protecting. Calculations or measurements documenting self-protection shall be approved by two licensed SROs.
2. Fission converter fuel elements shall be stored in accordance with the provisions of Specification 5.4.1 and Specification 5.4.2 or 5.4.3 as applicable.
3. Prior to transferring an irradiated element from the fission converter tank to the transfer cask, the operating history for the element shall be in compliance with any one of the following three requirements:

    a) Continuous operation at or below 50 kW for the four days prior to refueling.

    b) A maximum operating time of 4.8 hours per day at or below 250 kW during the four days prior to refueling.

    c) A maximum burnup of 436 kWh per fuel element during the four days prior to refueling.

Basis

Specification 5.4.4 requires that prior to transferring an irradiated element, that fuel element shall not have been operated in the reactor core at a power level above 100 kW for at least four days. This requirement can not be translated directly to the fission converter because of the different numbers of elements in the reactor core and in the fission converter. Alternatively, an equivalent power history is used for the fission converter.

A study was conducted in the SER to calculate the fuel plate temperature during fuel element removal. It was assumed that the fission converter was operated continuously at its maximum operating power of 250 kW until four days prior to removal of the fuel element. During those four days, operation was as described in the specification. It was also assumed that all heat transfer was by radiation alone during the fuel transfer. The maximum clad temperature was calculated to be 313°C which is well below the Al-6061 softening temperature of 450 °C.

6.6.2.4 D_2/H_2 Concentration and Recombiner Operation

Applicability

This specification applies to the D_2 or H_2 gas concentration in the helium cover gas blanket over the fission converter tank, and to the operation of the recombiner system. In the event that the fission converter is operated without its top shield lid, this specification is not applicable.

Objective

To prevent a flammable concentration of either D_2 or H_2 gas in the helium blanket.

Specification

1. The D_2 concentration in the helium blanket shall not exceed 6 volume percent if D_2O is used as the primary coolant in the fission converter.
2. The H_2 concentration in the helium blanket shall not exceed 6 volume percent if H_2O is used as the primary coolant in the fission converter.
3. The recombiner shall be operated for a minimum of 5 hours per month in any month during which the fission converter was operated.
4. In the event that the recombiner is not operable, fission converter operation may be continued provided that D_2/H_2 samples are taken weekly, and that the D_2/H_2 concentration in the helium blanket does not exceed 2 volume percent.

Basis

Recombination of the disassociated D_2/H_2 and O_2 is accomplished by circulating the helium from above the fission converter tank through a recombiner.

The concentration limit of D_2/H_2 in helium blanket is obtained from Specification 3.3.1, in which the concentration is conservatively determined from extrapolation of flammability limits.

6.6.2.5 Fission Converter Safety System

Applicability

This specification applies to the operability of the fission converter safety channels.

Objective

To ensure that adequate automatic protective actions are provided by the safety channels during operation of the fission converter.

Specification

1. The fission converter shall not be operated unless the safety channels listed in Table 6.6.2.5-1 are operable.
2. Emergency power with the capacity to operate the equipment listed in Table 6.6.2.5-2 of this specification shall be available whenever the fission converter is operating and shall be capable of operation for at least one hour following a loss of normal power to the facility.
3. There shall be an alarm at 110% or less of the fission converter's nominal operating power for fission converter operation using forced convection. The fission converter's nominal operating power shall be determined in accordance with Specification 6.6.4.8. This alarm shall not exceed 275 kW.
4. There shall be a reactor scram at the reactor power corresponding to the fission converter power 20 kW or less for fission converter operation using natural convection.

Basis

The parameters listed in Table 6.6.2.5-1 are monitored by the fission converter safety system. This system automatically initiates converter control shutter closure and/or a reactor scram to ensure that the LSSS and safety limits are not exceeded.

The use of emergency power is not essential for the fission converter because loss of power automatically scrams the reactor and thus the fission converter. The water shutter closes by gravity upon power failure. Nevertheless, the information supplied to the reactor operator and fission converter user that the fission converter is shut down will ensure personnel radiation safety. The choice of a minimum of one hour is based on Specification 3.6.1.

For forced convection cooling, protection against a fission converter overpower condition is provided by an alarm at 110% of nominal operating power and an automatic CCS closure at the over-power setpoint 275 kW. A reactor scram on fission converter overpower is not needed because the reactor itself will have already scrammed on high power. For natural convection cooling, protection against a fission converter overpower condition is provided by a reactor scram at the reactor power corresponding to the fission converter power 20 kW or less for fission converter operation using natural convection. This different approach is necessary because an overpower condition can occur on the fission converter during natural convection cooling even though the reactor itself is operating within its licensed operating power.

Table 6.6.2.5-1 Minimum Required Safety Channels for Fission Converter Operation

Channel	Automatic Action	Setpoint Range	Min. No. Required
Operation with Forced Convection Flow			
Primary Flow Rate	Reactor Scram* and Converter Control Shutter Closure	≥ 45 gpm	1
Power	Converter Control Shutter Closure	≤ 300 kW	1
Outlet Temperature	Converter Control Shutter Closure	≤ 60 °C	1
Coolant Level	Reactor Scram* and Converter Control Shutter Closure	≥ 2.1 m	1
Manual Reactor Minor Scram from the Fission Converter Medical Control Panel	Reactor Scram*	N/A	1
Operation without Forced Convection Flow			
Power	Reactor Scram ** and Converter Control Shutter Closure	≤ 20 kW	1
Outlet Temperature	Converter Control Shutter Closure	≤ 60 °C	1
Coolant Level	Reactor Scram* and Converter Control Shutter Closure	≥ 2.5 m	1
Manual Reactor Minor Scram from the Fission Converter Medical Control Panel	Reactor Scram*	N/A	1

* Not required if fission converter is in either a shutdown or a secured condition.

** For natural convection operation only and not required if fission converter is in either a shutdown or a secured condition.

Table 6.6.2.5-2 Minimum Equipment to be Supplied by Emergency Power

1.	Fission converter medical therapy room radiation monitor.
2.	Intercom between the fission converter medical therapy room and its associated medical control panel area.
3.	Intercom between the fission converter medical control panel area and the reactor control room.
4.	Emergency lighting of the fission converter medical therapy room and its associated medical control panel area.
5.	Outlet temperature and coolant level channels listed in Table 6.6.2.5-1.

6.6.2.6 Fission Converter Primary Coolant Quality Requirements

Applicability

This specification applies to the pH, conductivity, and activity of the fission converter primary coolant.

Objective

To control corrosion of the fission converter fuel and primary coolant loop structure, and activation of impurities and leakage of fission products in the fission converter primary coolant.

Specification

1. The pH of the fission converter primary coolant shall be kept between 5.5 and 7.5, except as noted in provision (4) below.
2. The conductivity of the fission converter primary coolant shall be kept less than 5 μS/cm at 20°C, except as noted in provision (4) below.
3. Any gross β-γ sample activity that exceeds the average of the previous monthly values (normalized by power) by a factor of three or more shall be investigated to determine the cause.
4. Operation of the fission converter with the pH or conductivity outside the limits given in (1) and (2) above is permitted provided:

   a) the pH is between 5.0 and 8.0,

   b) any increase in conductivity is not the result of a chloride ion concentration in excess of 5 ppm,

   c) sampling of the fission converter coolant is done at least once every eight hours, and

   d) the pH band specified in provision (1) is re-established within 48 hours.

   Otherwise, the fission converter shall not be operated.

Basis

The purpose of pH monitoring is to ensure corrosion on the fission converter fuel and the primary coolant loop structure is maintained within an acceptable limit. The fission converter fuel cladding and the fission converter tank are made of aluminum alloy. A portion of the primary coolant loop is constructed of stainless steel. Lower pH will reduce aluminum alloy corrosion and oxide film formation on the fuel surface and higher pH is favored to control stainless steel corrosion. Thus a pH range between 5.5 and 7.5 is selected for the fission converter primary coolant.

Electrical conductivity is also monitored to control purity of the fission converter primary coolant. A conductivity limit of 5 μS/cm has been traditionally adopted by research reactors.

The criterion that gross β-γ activity three times in excess of the average value be investigated is in accordance with industry practice for the detection of incipient fuel failure. In order for this criterion to be applied with a consistent basis, only samples that have similar power histories should be compared.

Operation with out-of-specification chemistry is acceptable for short intervals. The important factors are pH and the absence of a high chloride concentration. A high conductivity by itself is not of concern.

6.6.3 Fission Converter Surveillance Requirements

Applicability

This specification applies to the surveillance of safety systems whose operation is important to fission converter safety.

Objective

To ensure the reliability of the instrumentation important for safe operation of the fission converter.

Specification

1. The following instruments or channels for the fission converter safety system will be tested at least monthly and each time before startup of the reactor if the reactor has been shut down more than 24 hours and if the fission converter facility will be used within that reactor operating period. The monthly requirement may be omitted if the fission converter facility will not be used during that month.

Instrument, Channel, or Interlock	**Functional Test**
Primary coolant flow	Automatic converter control shutter closure and reactor scram
Power level	Automatic converter control shutter closure
Primary coolant outlet temperature	Automatic converter control shutter closure
Fission converter tank coolant level	Automatic converter control shutter closure and reactor scram

2. The following instruments used in the fission converter facility shall be calibrated and trip points verified when initially installed, any time the instrument has been repaired, and at least annually:
   a) Neutron flux level channel,
   b) Primary coolant flow channel, and
   c) Primary coolant outlet temperature channel.

3. The neutron flux level channel and a fission converter primary system heat balance shall be checked against each other at least annually and when design changes in the reactor and/or the fission converter are made that may affect the existing calibration result.
4. The gross β-γ activity of the fission converter primary coolant shall be determined at least monthly. The conductivity of the fission converter primary coolant shall be determined either by a continuous on-line instrument or a monthly sample. The pH of the fission converter primary coolant shall be measured monthly if the average conductivity exceeds 0.10 μS/cm if H_2O is used as a coolant or 0.03 μS/cm if D_2O is used as a coolant. The tritium content of the coolant shall be determined quarterly if D_2O is used as the fission converter primary coolant.
5. The following instruments used in the fission converter shall be subject to a functional test when initially installed, any time that the instrument has been repaired, and at least annually:

   Fission converter tank coolant level channel.

Basis

The specifications for functional tests, calibrations, and primary coolant sampling adhere to current MITR-II practice.

The annual frequency for performance of the calorimetric was chosen because the fission converter's power is a function of the MITR-II's power and the burnup of the fission converter fuel. The latter will occur very slowly. Hence, the annual performance of a calorimetric is sufficient to detect any change in fission converter power production.

Experience with the MITR-II primary and D_2O systems has shown that an out-of-specification chemistry condition is extremely rare. Heat fluxes present in the fission converter are too low to contribute to fuel cladding degradation in the event of out-of-specification chemistry. Continued operation of the fission converter is thus permitted.

6.6.4 Fission Converter Design Features

Applicability

This specification applies to the design of the fission converter tank, fueled region, and primary coolant system.

Objective

To ensure compatibility of the fission converter design features with the present safety evaluation.

Specification

1. The fission converter primary coolant system can utilize either H_2O or D_2O coolant.
2. All materials that are in contact with primary coolant, including those of the converter tank, shall be aluminum alloys, stainless steel, or other materials that are chemically compatible with H_2O and D_2O coolant, except for small non-corrosive components such as gaskets, filters, and valve diaphragms.
3. The fueled region of the fission converter may consist of up to eleven fuel elements of a type described in Specification 5.3.1.
4. The fueled region of the fission converter may contain sample assemblies provided that they conform to the requirements of Specification 6.6.2.1.4. Design of the sample assemblies shall also conform to the following criteria:

    a) They shall be secured either by a mechanical device or by gravity to prevent movement during fission converter operation,

    b) Materials of construction shall be radiation resistant and compatible with those used in the fission converter fueled region and primary coolant system,

c) Sufficient cooling shall be provided to ensure structural integrity of the assembly and to preclude any boiling of the primary coolant, and

d) The size of the irradiation thimble shall be less than 16 square inches in cross section.

5. The removable aluminum block shall be installed in the fission converter tank unless calculations to show compliance with Specification 6.6.2.1.1, 6.6.2.1.2, and 6.6.2.1.6 have been performed for another configuration. Other configurations could include but are not limited to a block of a different material, the absence of the block, or a combination of a solid material and coolant.

6. The pumps and other components of the fission converter's primary cooling system shall be located so as to prevent uncovering of the fission converter fuel elements as a result of siphoning.

7. The following interlocks shall be installed in order to prevent fission converter operations under abnormal conditions:

   a) Interlock that prevents opening of the converter control shutter without the fission converter primary flow scram operable (forced convection operation only).

   b) Interlock that prevents opening of the converter control shutter without the fission converter coolant level scram operable.

   c) Interlock that automatically closes the water and mechanical shutters when the medical room control panel key switch is turned to the OFF position.

   d) Interlock that ensures the CCS will close automatically when the CCS control panel key switch is in the OFF position.

   e) Interlock that prevents startup of the MIT Research Reactor unless the CCS is in the fully closed position.

8. The fission converter’s nominal operating power for the given combination of MITR-II licensed power, fission converter coolant (H_2O or D_2O), and U-235

content of the fission converter fuel shall be estimated prior to initial use and confirmed during initial use.

Basis

The two materials specifications are in keeping with those imposed on the design of the MITR-II. The specifications on fuel type and sample assemblies impose the same criteria as used for the MITR-II.

Design criteria for the sample assemblies were adopted from the criteria for the MITR-II in-core sample assemblies Specification 5.3.2.

Removing or replacing the movable aluminum block with other material may cause a different power distribution as well as changing k_{eff}. Therefore, any change in configuration shall be evaluated to show compliance with existing technical specifications.

6.6.5 Reporting Requirements

Applicability

This specification applies to the reporting requirements and the contents of the initial startup tests of the fission converter system.

Objective

To ensure that adequate management controls are available for safe operation of the fission converter.

Specification

1. A written report to the Document Control Desk, USNRC, Washington, D.C shall be made within 90 days after completion of the startup testing of the fission converter that is required upon receipt of a new facility license or an amendment to the license authorizing an increase in fission converter power level. This report shall describe the measured values of the operating conditions or characteristics of the reactor under the new conditions, including:

   a) An evaluation of facility performance to date in comparison with design predictions and specifications; and

   b) A reassessment of the safety evaluation submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior evaluation.

2. The startup report shall include the following items:

   a) calculation of k-effective for the initial fuel loading,

   b) measurements and comparison to prediction of flow disparity,

   c) determination and comparisons to prediction of nuclear hot channel factor, and

   d) fission converter power measurements and calibrations.

6.7 Experiments Involving In-Core Irradiation of Fissile Materials

Applicability

This specification applies to the in-core irradiation of fissile materials. It does not apply to out-of-core irradiations.

Objective

To ensure that fissile materials experiments do not affect safe operation of the reactor and to provide for the protection of the public health and safety by ensuring the integrity of irradiated fissile materials.

Specification

1. In-core fissile materials irradiation experiments shall not contain circulating loops.
2. The physical form of the fissile materials shall be solid. The fissile materials shall be doubly encapsulated to preclude radionuclide leakage during irradiation.
3. The cross-section of an in-core fissile materials experiment facility shall not exceed 16 square inches.
4. The total initial amount of U-235 in each in-core fissile materials experiment shall not exceed 100 grams. Any mixture of fissile materials is permitted provided that the off-site dose consequences are less than those of 100 grams of U-235.
5. Thermal power generated from each fissile materials experiment shall not exceed 100 kW during irradiation.
6. Each fissile materials irradiation experiment shall be monitored so that over-temperature protection is provided by an automatic reactor scram. The automatic reactor scram function will be tested each time before startup of the reactor if the

reactor has been shutdown for more than 24 hours. The temperature channels shall be calibrated and trip points verified when initially installed, any time the instrument has been repaired, and at least annually.

7. Any void space between the inner and outer barriers of the double encapsulation shall be sampled at least weekly for indication of fission products during any week that the experiment is in core and the reactor power exceeds 100 kW. The finding shall be compared to a baseline and the reactor power shall be made less than 100 kW if fission product activity exceeds three times baseline.

8. Design of the fissile materials experiment shall conform to the provisions of TS#6.1, "General Experiment Criteria." Each proposed in-core fissile material's experiment shall require a documented safety review and approval by the MIT Reactor Safeguards Committee (MITRSC) or, if authorized by the MITRSC, by its Subcommittee for in-core experiments.

Basis

The MITR-II is licensed as a research reactor. Code of Federal Regulations 10 Part 50.2 defines a non-power reactor as a research or test reactor licensed under 10 CFR 50.21(c) or 50.22 for research and development. A test facility is defined in 10 CFR 50.2 as a nuclear reactor for which "…an application has been filed for a license authorizing operation at: (1) a thermal power level in excess of 10 megawatts; or (2) a thermal power level in excess of 1 megawatt, if the reactor is to contain: (i) a circulating loop through the core in which the applicant proposes to conduct fuel experiments; or (ii) a liquid fuel loading; or (iii) an experimental facility in the core in excess of 16 square inches in cross-section." Therefore, Technical Specifications 6.7.1, 6.7.2, and 6.7.3 are based on 50.2(2)(i), 50.2(2)(ii), and 50.2(2)(iii), respectively.

Other limits on the in-core irradiation of fissile materials specific to the MITR-II in-core experiments are experiment reactivity worth limit and onset of nucleate boiling (ONB). Additional requirements such as weekly sampling of cover gas in the void space and over-temperature automatic reactor scram provide redundant protection against a potential malfunction of the fissile materials irradiation experiments. The limit on U-235 content in a fissile materials irradiation experiment is derived from the Design Basis Accident (DBA) of the reactor. The effect of actinides, which are produced from U-238, on off-site dose is analyzed. It is concluded that a limit on the initial amount of U-238 is not required.

The Design Basis Accident (DBA) chosen for the MITR-II assumes a blockage of five coolant flow channels that results in four plates completely melted [6.7-1]. Release of the fission products to the atmosphere is calculated assuming that the fission product buildup achieved saturation. Off-site dose to the general public is then calculated from the released fission products. The maximum amount of fissile materials that can be accommodated in a fissile materials experiment should result in a maximum fission product release below that of the DBA. Using an approximation based on the U-235 content, the maximum amount of U-235 would be 506 grams (mass of U-235 per fuel element) x 4 (plates) $\div$ 15 (plates per element) = 135 grams. A limit of the total initial amount of 100 grams U-235 is conservatively chosen.

Actinides are produced when U-238 is irradiated. The off-site whole body dose from actinides was calculated to be 2 mrem per kilogram of initial U-238 [6.7-2]. The maximum initial amount of U-238, which is set by the total off-site dose from both fission products and actinides releases of the fissile materials experiment, was calculated to be 31 kg. This amount is significantly higher than that of natural uranium that contains 100 grams of U-235, 0.1 kg U-235 x (0.993/0.007) = 14.2 kg U-238. Therefore, a limit on the initial amount of U-238 is not required.

The limit on the thermal power generated from the fissile materials experiment is primarily imposed by the onset of nucleate boiling (ONB), which is one of the criteria in TS #6.1. Each in-core fissile materials experiment design will be reviewed to ensure that ONB would not occur during steady-state operation. However, 100 kW, which is less than the average thermal power per fuel element of 200 kW, is used to set an upper bound for any fissile materials irradiation experiment.

The inner barrier of the double encapsulation is monitored for over-temperature. The scram setpoint is experiment-dependent and will be chosen to avoid rapid mechanical and/or chemical degradation of the barrier. The scram setpoint will be documented in the safety review that is required by provision 8 of this technical specification.

Fission product gas release is an unlikely accident scenario. An analysis is performed assuming that, as a result of multiple failures, the entire fission product inventory produced from a fissile materials experiment is released through the reactor ventilation system. The fission product gases analyzed here are the noble gas nuclides including Kr-85m, Kr-87. Kr-88, Xe-131m, Xe-133, and Xe-135. The fission product gas inventory is assumed to be at equilibrium at 100 kW (maximum allowable power of a fissile materials irradiation experiment) and is released within one week. The interval of one week is chosen because that is the frequency for the void space sampling. This is a conservative assumption because a shorter interval will result in a much higher core purge monitor reading and hence increase the probability of detection. The analysis concludes that (a) the core purge monitor should detect a higher reading, 44 kcpm over background, if the fission product gases were to escape the barriers, an increase equivalent to approximately twice that of a normal background reading, and (b) if the entire fission product gas inventory leaks from a fissile materials irradiation experiment and is released to the atmosphere through the stack, the total inhalation dose is calculated to be about 17 mrem. The inhalation dose of 17 mrem is much lower than the 100 mrem annual limit for the general public defined by 10 CFR 20. There will be no additional thyroid dose because none of the fission product gases affect the thyroid.

References

6.7-1 MITR-II Staff, "Safety Analysis Report for the MIT Research Reactor (MITR-II)," Report No. MITNE-115, 22 Oct. 1970.

6.7-2 File Memo, "Actinides Off-Site Dose Calculations During DBA," Oct. 2001.

6.7-3 File Memo, "Off-Site Dose Calculations for Fission Product Gases Release," August 2002.

7. ADMINISTRATIVE CONTROLS

Applicability

Administrative controls are the means by which reactor operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, reactor organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of the measures is applicable as minimum requirements throughout reactor life.

Objective

To ensure that adequate management controls are available for safe facility operation.

7.1 Organization

7.1.1 Structure

The organization for the management and operation of the reactor facility is shown in Figure 7.1-1.

7.1.2 Responsibility

1. The Director of Reactor Operations is directly responsible for the safe operation of the facility.

2. In all matters pertaining to safe operation of the MIT Reactor (MITR-II) and to these Technical Specifications, the Director of Reactor Operations shall report to and be directly responsible to the Managing Director for Operations at the Nuclear Reactor Laboratory. The management organization is shown in Figure 7.1-1.

3. The MIT Reactor Radiation Protection Officer shall be responsible for radiation protection at the MITR-II. He shall advise the Director of Reactor Operations on all matters pertaining to radiation protection.

4. The MIT Reactor Radiation Protection Officer shall report to and be directly responsible to the Director of MIT Environment, Health, and Safety Office.

5. The MIT Reactor Radiation Protection Officer shall be a member of the MIT Reactor Safeguards Committee

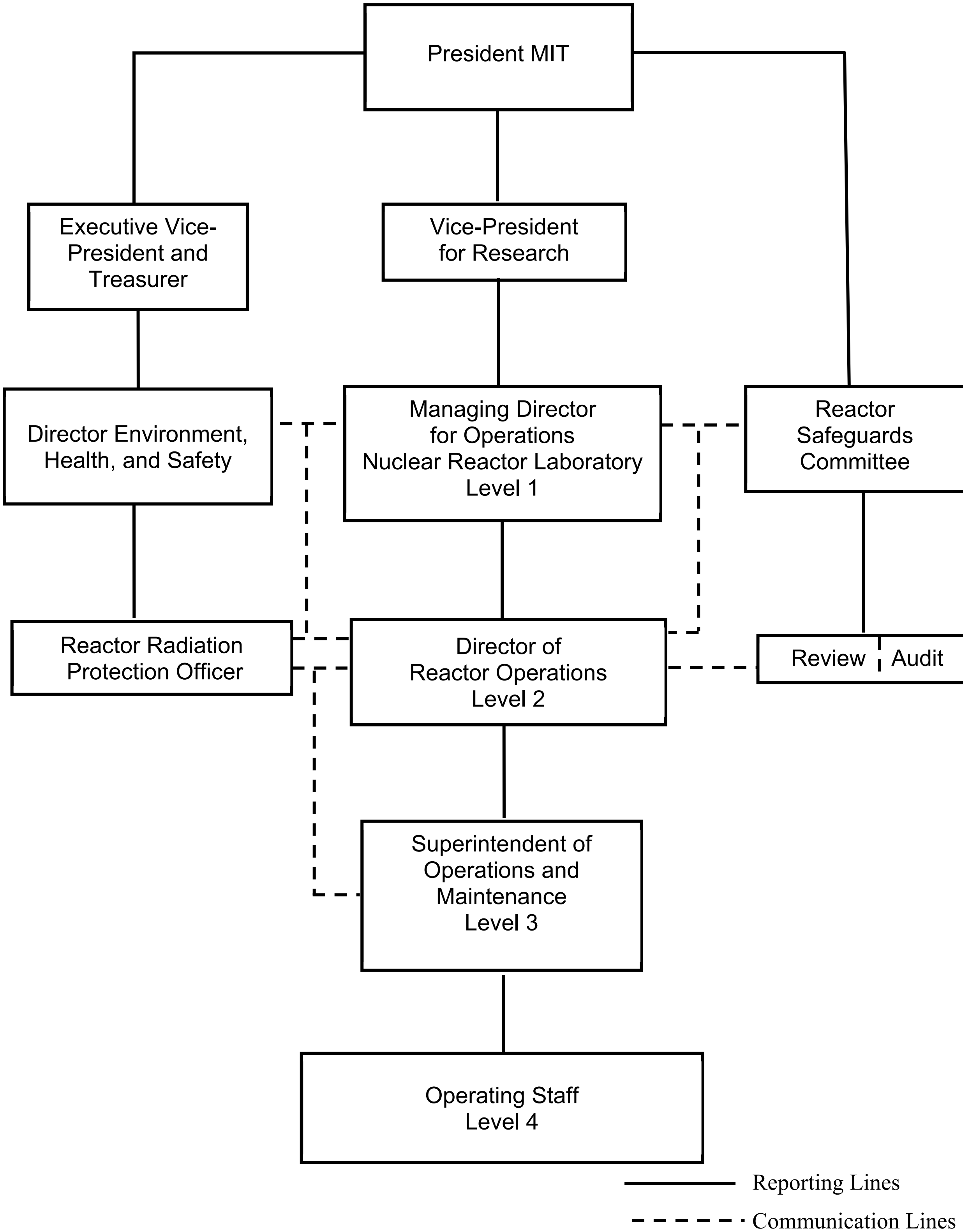

Figure 7.1-1: Management Organization for Reactor Operations

6. In the event of disagreement between the recommendations of the MIT Reactor Radiation Protection Officer and the Director of Reactor Operations or their alternates, on matters pertaining to radiation protection, the course determined by the Director of Reactor Operations or his designated alternate to be the more conservative shall be followed. Records of the disagreement shall be sent for review and possible reconsideration to the Director of MIT Environment, Health, and Safety Office, the Managing Director for Operations at the Nuclear Reactor Laboratory, and the Chairman of the MIT Reactor Safeguards Committee.

7. The responsibilities of any given level in the management organization chart may be assumed by designated alternates conditional upon appropriate qualifications. Such delegation of authority shall be documented in writing by the regularly assigned individual or that person's immediate supervisor.

7.1.3 Staffing

The minimum reactor staff organization shall be as follows:

1. When the reactor is not shut down, the minimum crew complement for a shift shall be two licensed operators including at least one licensed senior reactor operator, one of whom shall be in the control room. In addition, the MITR-II Radiation Protection Officer or a designated alternate shall be onsite or on call. If on call, one of the licensed operators will have responsibility for implementing radiation protection procedures.

2. Whenever the reactor is not secured, two persons shall be onsite, one of whom shall be a licensed senior reactor operator. An operator or senior operator

shall be present in the control room. In addition, the MITR-II Radiation Protection Officer or a designated alternate shall be onsite or on call. If on call, the onsite licensed senior operator will have responsibility for implementing radiation protection procedures.

3. A list of reactor facility personnel by name and telephone number shall be readily available in the control room. This list shall include management, radiation safety, and other operating personnel.

7.1.4 Selection of Personnel

Minimum educational and/or experience requirements for those individuals who have line responsibility and/or authority for the safe operation of the facility are as follows:

1. Director of Reactor Operations - The Director of Reactor Operations shall have a minimum of seven years of nuclear experience. The individual shall have a recognized baccalaureate or higher degree in an engineering or scientific field. Education or experience that is job-related may be substituted for a degree on a case-by-case basis. The degree may fulfill four years of the seven years of nuclear experience required on a one-for-one time basis. At least three years of experience shall be in a responsible position in reactor operations or a related field including at least one year's experience in reactor facility management or supervision. The Director of Reactor Operations shall hold a senior operator's license for the MIT Research Reactor, or have held the equivalent at the MIT Research Reactor or another reactor facility. In the case of the latter, the individual shall receive facility-specific training, based on the individual's background and abilities.

2. Superintendent of Operations and Maintenance - The Superintendent of Operations and Maintenance shall have a minimum of five years of responsible reactor experience. A maximum of two years of experience may be fulfilled by academic or related technical training on a one-for-one basis. The Superintendent shall hold a senior operator's license for the MIT Research Reactor.

3. Reactor Radiation Protection Officer - The Reactor Radiation Protection Officer shall have a minimum of five years of experience in radiation protection including at least one year of experience at a nuclear reactor

facility. A maximum of four years of the five years experience may be fulfilled by related technical or academic training.

4. Shift Supervisor - Shift supervisors shall have a minimum of a high school diploma or equivalent and three years of responsible reactor experience. A maximum of 1-1/2 years experience may be fulfilled by academic or related technical training on a one-for-one basis. The requirement for experience may be completely satisfied by an advanced degree in Nuclear Engineering. Shift supervisors shall hold an NRC senior operator license for the MIT Research Reactor.

5. Reactor Operator - Reactor operators shall have a high school diploma or equivalent. They shall hold an NRC reactor operator license for the MIT Research Reactor.

7.1.5 Training of Personnel

1. A training program shall be established to maintain the overall proficiency of the Reactor Operations organization. This program shall include components for both initial certification and requalification.

2. The training program shall be under the direction of the Superintendent of Operations and Maintenance and/or the Director of Reactor Operations.

3. Records of individual plant staff members' qualifications, experience, training, and requalification shall be maintained as specified in Specification 7.8.2.

7.2 Review and Audit

7.2.1 MIT Reactor Safeguards Committee

Overall direction on matters of reactor safety rests with the MIT Reactor Safeguards Committee or MITRSC. Approval of the MITRSC is necessary for all new operating plans and policies and all significant modifications thereto which may involve questions of nuclear safety. The MITRSC is also responsible for auditing operation of the reactor. The Chairman of the MITRSC reports directly to the President of MIT. The MITRSC communicates directly with the Managing Director for Operations at the Nuclear Reactor Laboratory and with the Director of Reactor Operations, both of whom are members of the MITRSC.

1. Composition and Qualifications: The MITRSC shall be composed of a minimum of nine persons with not more than one-third of the total membership chosen from the reactor staff organization and a minimum of three members from outside MIT. All members and the Chairman shall be selected by the President of MIT. At least four voting members including participating alternates shall have a minimum of a Bachelor's Degree in Engineering or the Physical Sciences and have a minimum of three years of professional level experience in nuclear services, nuclear plant operation, or nuclear engineering, and the necessary overall nuclear background to determine when to contact consultants for analyses beyond the scope of the MITRSC's expertise. Ex-officio members shall include the MIT Radiation Protection Officer and a representative of the MIT Environment, Health, and Safety Office.

2. Charter and Rules

   a) Meeting Frequency: Meetings shall be held at least annually.

   b) Quorum: A quorum shall consist of at least a majority of the Committee’s voting membership. In addition, either the Chairman or a designated alternate shall be present. Finally, no more than a minority of the quorum shall have line responsibility for reactor operations.

c) Subcommittee: The full MITRSC may, after discussion of a particular topic, designate a subcommittee to conduct further review. The MITRSC may also choose to delegate approval authority to a subcommittee. The membership of a subcommittee, its chairman, its purpose, and its authority must be stated in the minutes of a meeting of the full MITRSC. These items must be reaffirmed at least annually. The quorum rule that applies to the membership of the MITRSC as a whole also applies to the membership of any subcommittee.

d) Minutes: Minutes are kept of all MITRSC (and subcommittee) meetings. These are distributed to all members of the Committee, to the MIT Administration, and to the Reactor Operations/RRPO Staff.

e) Access to Reactor Records: The MITRSC has complete access to all records pertaining to reactor operation.

7.2.2 MITRSC Review Function

1. The MITRSC's review function shall include the following:

   a) Determinations that changes in equipment, systems, tests, experiments, or procedures described in the annual report pursuant to Specifications 7.7.1.4 and 7.7.1.5 do or do not meet the criteria of 10 CFR 50.59(c)(1).

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

   c) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

   d) Proposed changes in technical specifications or license.

   e) Violations of technical specifications or license as well as violations of internal procedures or instructions having safety significance.

   f) Operating abnormalities having safety significance.

   g) Reportable occurrences.

   h) Audit reports.

2. The review findings shall be documented in the MITRSC minutes.

7.2.3 MITRSC Audit Function

1. The MITRSC audit function may be performed by members of the MITRSC who do not have line responsibility for the reactor or by a consultant who has qualifications equivalent to those listed in Specification 7.2.1.1

2. Audits shall be performed at least annually.

3. The scope of the audit shall include, as a minimum, the following:

   a) Facility operations for conformance to the technical specifications and license conditions.

   b) The requalification program for the operating staff.

   c) The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety.

   d) The reactor facility emergency plan and implementing procedures.

   e) The physical security plan.

   Deficiencies uncovered that affect reactor safety shall immediately be reported to the Managing Director for Operations at the Nuclear Reactor Laboratory. A written report of the findings of the audit shall be submitted to the Managing Director for Operations at the Nuclear Reactor Laboratory and to all MITRSC members within three months after the audit has been completed.

7.3 Radiation Safety

1. The Radiation Protection Program shall be designed to achieve the requirements of 10 CFR 20.

2. An ALARA program shall be designed that applies to facility staff and users as well as to the general public and the environment.

3. The Reactor Radiation Protection Officer has the authority to interdict or terminate activities that may compromise safety. Disagreements between Reactor Operations and Reactor Radiation Protection shall be resolved pursuant to Specification 7.1.2.6.

7.4 Procedures

7.4.1 Review Process

Prior to the implementation of a new procedure or a change to an existing one, a written safety review shall be prepared that includes the following:

1. A description of the new procedure and/or change to an existing one shall be summarized.
2. A safety evaluation of the new procedure and/or change to an existing one shall be prepared.
3. The new procedure and/or change shall be evaluated to determine if:
    a) The criteria of 10 CFR 50.59 (c)(1) are not met, or
    b) The emergency or security plans are affected, or
    c) The requalification program is affected, or
    d) The ALARA program is impacted.

7.4.2 Approval Process

No new procedure or a change to an existing one shall be implemented until the material prepared in Specification 7.4.1 has been reviewed and approved by two licensed senior reactor operators and the Director of Reactor Operations. Where radiation protection considerations are involved, the review and approval of the MITR-II Radiation Protection Officer, or a designated alternate, shall also be required. Review and approval by the MITRSC is also required for procedures that concern:

a) The standard operating plan for the reactor including startup and shutdown procedures.

b) The emergency plan and its implementing procedures.

c) The security plan and its implementing procedures.

d) The requalification program.

In the event that the review required by Specification 7.4.1 identifies a situation in which the criteria of 10 CFR 50.59 (c)(1) are not met, or finds that the emergency or security plan is degraded or that the requalification program is weakened or that the ALARA program is negatively impacted, then the proposal must be referred to the MITRSC (and possibly to the NRC).

7.4.3 Scope of Procedures

Written procedures that have been reviewed and approved pursuant to Specification 7.4.1 and 7.4.2 shall be prepared for all operational activities described in the Safety Analysis Report. As a minimum, these include:

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

b) Refueling operations.

c) Maintenance of components that have nuclear safety significance.

d) Surveillance and testing as required by these technical specifications.

e) Personnel radiation protection consistent with applicable regulations or guidelines. These procedures shall include a management commitment and programs to maintain experiments and releases in accordance with the ALARA concept.

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

g) Implementation of required plans such as emergency or security plans.

h) Use, receipt, and transfer of byproduct material.

7.4.4 Equipment Changes

The review and approval process (Specifications 7.4.1 and 7.4.2) shall apply to changes of equipment that can affect reactor safety.

7.5 Experiment Review and Approval

7.5.1 Review Process

Prior to performing any reactor experiment of a type not previously approved, the proposed experiment or irradiation series shall be evaluated in terms of its effect on reactor operation and the possibility and consequences of its failures, including, where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects. This evaluation shall normally consist of the following:

a) Preparation of an experiment description and safety evaluation by either the experimenter or individual(s) appointed by the Director of Reactor Operations.

b) Preparation and approval of a written safety review as described in Specifications 7.4.1 and 7.4.2.

c) Preparation of written procedures for the conduct of the experiment.

Item (a) above may be done as part of item (b) above at the discretion of the Director of Reactor Operations.

7.5.2 Approval Process

1. No reactor experiment of a type not previously approved shall be performed until the materials prepared in Specification 7.5.1 (a) and (b) have been reviewed and approved in writing by two licensed senior reactor operators, the Director of Reactor Operations, and the MITRSC. The materials prepared in Specification 7.5.1 (c) shall be reviewed and approved as required by Specification 7.4 (Procedures).

2. Substantive changes to previously reviewed experiments shall require the process described in Specifications 7.5.1 and 7.5.2.1. Minor changes that do

not significantly alter the experiment may be implemented upon the review and approval of a safety review by two senior reactor operators and the Director of Reactor Operations. The safety review shall, as a minimum, describe the change and evaluate its safety.

3. Previously approved experiments may be performed at the discretion of a licensed senior reactor operator without the necessity of any further review or approval.

4. All experiments shall conform to the General Experiment Criteria as given in Specification 6.1.

7.6 Required Action

7.6.1 Action to be Take in Case of Safety Limit Violation

The following action shall be taken in the event of a safety limit violation:

a) The reactor shall be shut down, and reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.

b) The safety limit violation shall be promptly reported to the Superintendent of Operations and Maintenance, the Director of Reactor Operations, the Managing Director for Operations at the Nuclear Reactor Laboratory, and the Chairman of the MITRSC.

c) The safety limit violation shall be reported to the U.S. Nuclear Regulatory Commission.

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

   (i) The time and date of the violation, reactor status at the time of the violation, and a description of the violation.

   (ii) Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

   (iii) Effect of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public.

   (iv) Corrective action to be taken to prevent recurrence.

   This report shall be reviewed by the MITRSC and submitted to the U.S. Nuclear Regulatory Commission. Any follow-up report shall also be submitted to the U. S. Nuclear Regulatory Commission before authorization is sought to resume operation of the reactor.

7.6.2 Action to be Take in the Event of a Reportable Occurrence

The following action shall be taken in the event of a reportable occurrence:

a) The reactor shall be shut down unless the cause for the occurrence has been identified and rectified immediately upon discovery or unless the occurrence has no immediate safety significance to the reactor, to the safety of reactor personnel, and to the safety of the public.

b) The Director of Reactor Operations shall be notified.

c) Operation of the reactor shall not be resumed until so authorized by the Director of Reactor Operations, or a designated alternate.

d) The U.S. Nuclear Regulatory Commission shall be notified.

e) A report shall be prepared that includes the time and date of the occurrence, reactor status at the time of the occurrence, a description of the occurrence, an evaluation of the cause of the occurrence, a record of the corrective action taken, and recommendations for appropriate action to prevent or reduce the probability of recurrence. This report shall be submitted to the U. S. Nuclear Regulatory Commission and it will be reviewed by the MITRSC no later than its next regularly scheduled meeting.

7.7 Reports

7.7.1 Annual Report

An annual or operating report shall be submitted to the U.S. Nuclear Regulatory Commission within ninety days following the 31st of December of each year. Its content is as follows:

1. A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics and operating procedures related to reactor safety occurring during the reporting period, as well as results of surveillance tests and inspections required by these Technical Specifications.

2. Tabulation showing the energy generated by the reactor, the number of hours the reactor was critical, and the cumulative total energy output since initial criticality.

3. The number of emergency shutdowns and inadvertent scrams, including the reasons therefore.

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

5. A description of each change to the facility or procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59 including a summary of the safety evaluation of each.

6. A description of any reactor-related environmental surveys performed by MIT personnel outside the facility.

7. A summary of radiation exposures received by facility personnel and experimenters, including the dates and time of significant exposures, and a summary of the results of radiation and contamination surveys performed within the facility.

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

   a) Liquid Waste (summarized on a monthly basis)

      (i) Total gross radioactivity released during the reporting period, excluding tritium.

      (ii) Total tritium radioactivity and average concentration released during the reporting period.

      (iii) Total radioactivity (beta/gamma) released for specific nuclides, if the gross beta radioactivity exceeds 1×10^{-5} µCi/ml at point of release, during the reporting period.

      (iv) Total volume of effluent water (including diluent) during periods of release.

   b) Gaseous Waste

      (i) Radioactivity of principal radionuclides discharged during the reporting period for

         - Gases

         - Particulates, with half lives greater than eight days.

      (ii) The effluent concentration limit used and the estimated activity discharged during the reporting period, by nuclide for principal radionuclides, based on representative isotopic analysis.

c) Solid Waste

(i) The total amount of solid waste packaged.

(ii) The total activity and type of activity involved.

(iii) The dates of shipment and disposition (if shipped offsite).

9. A summary of the use of the medical therapy facilities for human therapy.

a) Investigative Studies

(i) Nature and status of the studies.
(ii) Number of subjects involved.

b) Human Therapy

(i) Number of patients treated.
(ii) Type of cancer treated.

7.7.2 Reportable Occurrence Reports

1. A report shall be made by telephone or other communication systems to the U.S. Nuclear Regulatory Commission Headquarters Operations Center within 24 hours of:

a) Operation in violation of a safety limit.

b) Any release of radioactivity to unrestricted areas above permissible limits, whether or not the release resulted in property damage, personal injury, or exposure.

c) Any reportable occurrence as defined in Specification 1.3.32

d) Any significant variation of measured values from a corresponding predicted or previously measured value of safety-related operating characteristics.

2. A written report shall be provided as a follow-up to the verbal one within ten working days of the occurrence. This report shall provide the information required by Specification 7.6.2(e). The report shall be submitted to the NRC Document Control Desk.

7.7.3 Special Reports

Special reports, other than reportable occurrence reports, shall be provided in writing within 30 days to the NRC Document Control Desk in the event of:

a) Permanent change in the facility organization including Level 1 or Level 2 personnel as defined in Figure 7.1-1.

b) Significant changes in the transient or accident analysis as described in this Safety Analysis Report.

7.8 Records Retention

7.8.1 Five-Year Record Retention

The following records shall be retained for five years or for the life of the component involved if less than five years:

a) Records of normal reactor operation including power levels and periods of operation at each power level. (Note: Excludes retention of supporting documents such as checklists, log sheets, etc., which shall be retained for a period of at least one year.)

b) Records of principal maintenance activities including inspection, repair, substitution, or replacement of principal items of equipment pertaining to nuclear safety.

c) Records of reportable occurrences.

d) Records of surveillance activities that are required by these technical specifications.

e) Records of reactor facility radiation and contamination surveys.

f) Records of experiments performed with the reactor.

g) Records of fuel inventories, receipt, and shipments. (Note: Records of individual fuel element usage shall be retained until the element is returned to the U.S. Department of Energy.)

h) Records of changes made in the operating procedures.

i) Records of audit reports including both internal audits and those performed for or by the MITRSC.

7.8.2 Six-Year Record Retention

The following records shall be retained for six years:

a) Records of individual licensed staff members indicating qualifications, experience, training, and requalification. (Note: These are retained at all times that the individual is employed or until certification is renewed.)

7.8.3 Life of Facility

The following records shall be retained for the life of the facility:

a) Records of radioactivity in liquid and gaseous waste released to the environment.

b) Records of off-site environmental monitoring.

c) Records of radiation exposures of all plant personnel and others who enter radiation control areas.

d) Records and drawing changes reflecting plant design modifications made to systems and equipment described in the Safety Analysis Report.

e) Records of radioactive shipments including solid waste disposal.

f) Records of each review of (1) exceeding the safety limit; (2) the automatic safety system not functioning as required by the limiting safety system settings; (3) or any limiting condition for operation not being met.