

MIT NUCLEAR REACTOR LABORATORY

AN MIT INTERDEPARTMENTAL CENTER

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August 6, 2010

U.S. Nuclear Regulatory Commission
Attn: Document Control Room
Washington, DC 20555

Re: Massachusetts Institute of Technology; License No. R-37; Docket No. 50-20; Response to RAI (TAC No. MA 6084) dated 7/16/2010 and Other Related Information

Dear Sir or Madam:

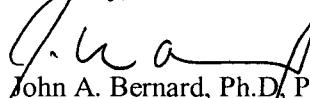
Attached please find the following information that is provided in support of the relicensing of the MIT Research Reactor.

- a) Response to RAI dated 07/16/2010.
- b) List of pages on which typographical errors were corrected.
- c) Protocol for initial power increase from 5 to 6 MW upon receipt of approval.
- d) Modification of TS 3.7 and TS 4.7 to allow use of an in-line tritium monitor.

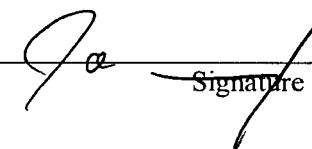
Enclosed is a revised copy of the proposed Technical Specifications.

Please contact the undersigned if there are any questions.

Sincerely,


John A. Bernard, Ph.D., PE, CHP
Director of Reactor Operations

I declare under the penalty of perjury that the foregoing is true and correct.

Executed on 6 Aug 10 
Date Signature

cc: William B. Kennedy
Project Manager
Research and Test Reactors Branch A
U.S. Nuclear Regulatory Commission

71020
NRK

Response to RAI Dated 07/16/10

1.	Requested change made.
2.	Proposed TS 2.1.1 has been revised to include the fuel cladding temperature as a safety limit.
3.	<p>The third paragraph of the basis of TS 3.3.6 is changed to read:</p> <p>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.</p> <p>A new fourth paragraph of the basis of TS 3.3.6 is added:</p> <p>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.</p>
4.	Requested change made.
5.	<p>The response to RAI TS 4.7.1 dated March 30, 2010 was confusing. Channel checks are performed two different ways. Prior to startup, they are accomplished using either an installed source or an electronic pulse. Once at power (>250 kW for 12 hours), they are accomplished using existing radiation field. Therefore, the following changes are made:</p> <ul style="list-style-type: none">• The first sentence of TS 4.7.1.1 is changed to read: “.....is operating above 250 kW for at least 12 hours.”• The second sentence of TS 4.7.1.1 is deleted because it was redundant with TS 4.7.1.2 which requires a channel test of the radiation monitors prior to startup if shutdown for more than 24 hours.
6.	Requested change made.
7.	Requested change made.
8.	A check of file memo No. 6.7-2 shows that the fourth line of the fourth paragraph of the basis should be 31 kg not 31 grams.
9.	Requested change made.

Typographical Errors

Typographical errors were corrected on the following pages:

Page #	Error
iii	Alignment
1-7	Spacing between paragraphs
1-9	No. instead of #
2-3	. instead of ,
3-17	Ensuring; or vs assuring; and
3-53	alert <u>the</u> operator
4-21	Alignment
6-62	ensure instead of insure
7-8	Office instead of Officer

Note: Page #s are those of Revision No. 5.

Protocol for Raising MITR Power from 5-6 MW

1.	Reactor systems (heat removal, nuclear instruments, shielding) verified operational for 6 MW.
2.	Raise reactor power to 5.0 MW (the original license limit) and establish conditions of both thermal and neutronic (including xenon) equilibrium.
3.	Raise reactor power to 5.5 MW. Verify that the response of the nuclear instrumentation is linear and that the heat removal and shield systems remain adequate. If not, lower power to 5.0 MW.
4.	Hold power at 5.5 MW until thermal and neutronic (including xenon) equilibrium are established. Re-verify conditions as per step 3).
5.	Raise reactor power to 6.0 MW. Verify that the response of the nuclear instrumentation is linear and that the heat removal and shield systems remain adequate. If not, lower power to 5.5 MW.
6.	Hold power at 6.0 MW until thermal and neutronic (including xenon) equilibrium are established. Re-verify conditions as per step 5.

In-Line Tritium Monitor

- TS 3.7.1.5(a) is modified to read:

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

- The seventh paragraph of the basis of TS 3.7.1 is changed to read:

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

- TS 4.7 is modified to include surveillance of the in-line tritium monitor (Note: It is understood that the requirement will only take effect once the in-line monitor is installed. It is on order but there is at least a 90 day lead time.)

Please see the attached memo, "Technical Basis for an In-line Tritium Monitor."

Reactor Radiation Protection Program

To: John Bernard, Director of Operations, NRL
From: Bill McCarthy, Reactor Radiation Protection Officer, EHS *WBM*
Date: August 3, 2010
Re: Technical basis for an In-line tritium monitor

The current system for monitoring the secondary water for radioactive materials involves the use of two inline gamma detectors and daily sampling of the secondary water for tritium analysis. We will soon be installing a new inline radiation monitor on the secondary system which is capable of analyzing for tritium as well as other beta and gamma emitters.



The sensitivity of the inline tritium monitor will be greater than the sensitivity for the current process of daily sampling and tritium analysis of the secondary water. An additional benefit is the timely detection of a leak of heavy water into the secondary system since it is monitoring in real-time. Other features of note are that it has built-in gamma spectroscopy, built-in check source for the beta gamma detector, and fault alarms for any signal loss from detectors.

A detailed analysis of the current radiation monitoring capabilities for the secondary system is covered in the letter January 24, 2004 in response to RAI dated May 30, 2001. This same analysis has been done for the inline tritium monitor and the results are outlined below.

H-3 activity released in one year due to an undetected leak:

The minimum detectable tritium activity in the secondary water will change from 4.5×10^{-5} uCi/mL to 5×10^{-6} uCi/mL. So the total theoretical undetected release change from 1.0×10^6 uCi via blowdown and 2.4×10^6 uCi via evaporation to 0.1×10^6 uCi via blowdown and 0.3×10^6 uCi via evaporation with the new system.

H-3 activity released in one year due to an undetected leak:

The great advantage of the inline monitor is real-time analysis of the tritium levels in the secondary water. The system's sensitivity to detect a leak increases with time (i.e. longer counting time yields better statistics). The published sensitivities range from 1.0 uCi/L in 20 minutes down to 0.005 uCi/L after seven days of sample processing (flow rate is 10 liters per minutes). The current system has a 24 hour leak detection timeframe. A large release could be detected in as little as 2 minutes. For the purpose of this study 20 minutes was used and a tritium level twice the sensitivity (2 uCi/L). The prior calculation used 24 hours and 1100 uCi/L. The total theoretical release changes from 5.6×10^7 uCi via blowdown and 1.3×10^8 uCi via evaporation to 1.3×10^5 uCi via blowdown and 3×10^5 uCi via evaporation with the new system.

Na-24 activity released in one year due to an detected and/or undetected leak:

No changes to this radiation monitoring system

**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.

1.3.27 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.

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.

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:

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

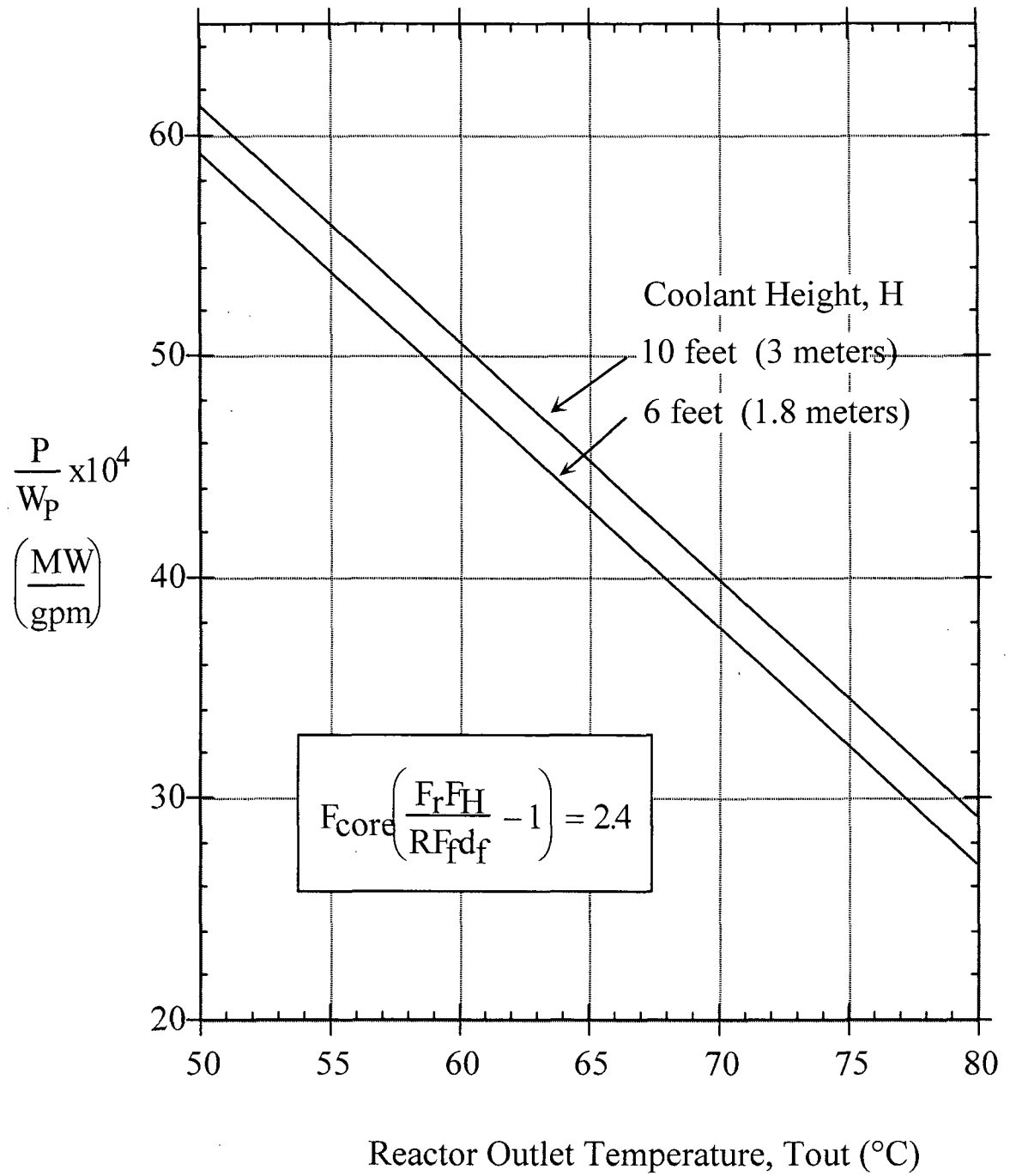


Figure 2.1-1 MITR-II Safety Limits for Forced Convection Operation

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:

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

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 ($^{\circ}\text{C}$), and

T_{out} is the average core outlet temperature ($^{\circ}\text{C}$).

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 $2.353 \times 10^4 \text{ W/m}^2$, 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:

$$T_{\text{out}} < 107 + \frac{P \cdot F_{\text{core}}}{W_p c_{\text{pf}}} + 0.01777(q''(z))^{0.466} - \frac{F_H}{\dot{m} c_{\text{pf}}} \int_0^z P_H q''(z) dz - F_{\Delta T} \frac{q''(z)}{h} \quad (2.2-1)$$

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:

$$T_{\text{out}} < 107 + \frac{PF_{\text{core}}}{W_p c_{\text{pf}}} + 0.01777 q''_{\text{avg}}{}^{0.466} - \frac{F_H F_r P}{\dot{m} c_{\text{pf}} N_c} - F_{\Delta T} \frac{q''_{\text{avg}}}{h} \quad (2.2-2)$$

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.

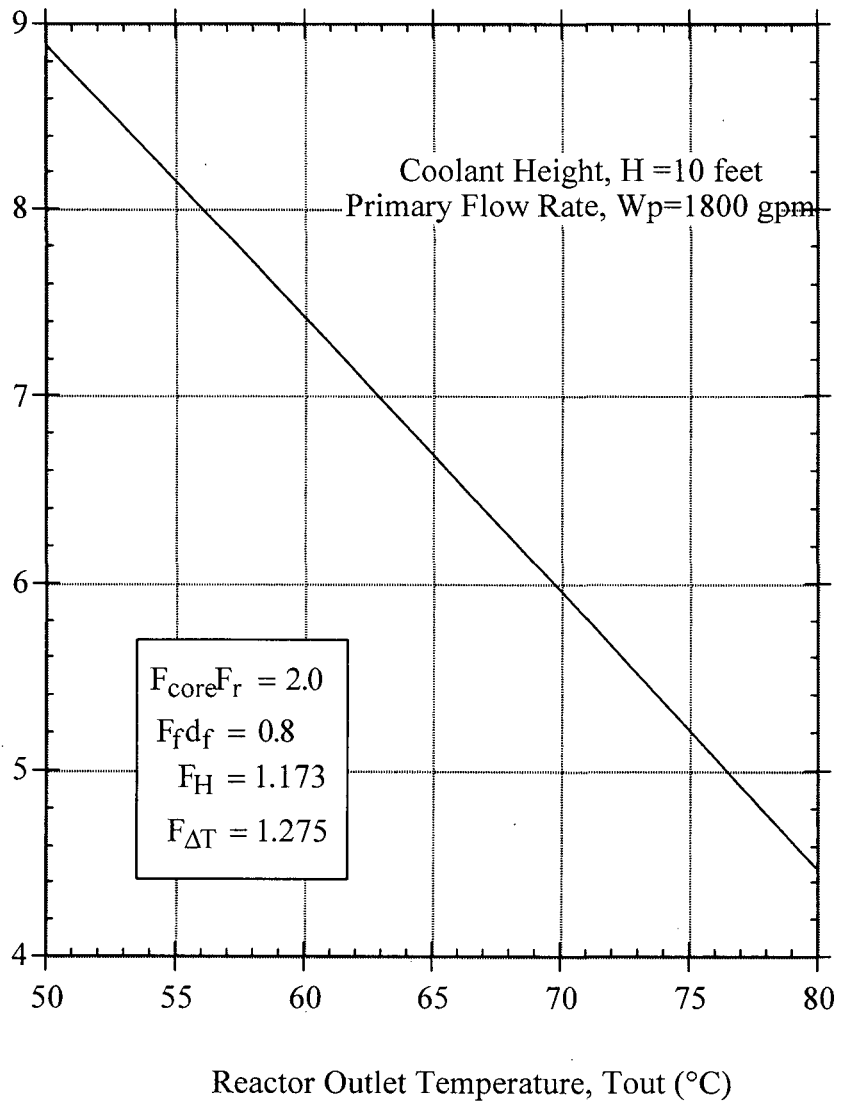


Figure 2.2-1 MITR-II Limiting Safety System Settings for Forced Convection Operation (Two Pumps).

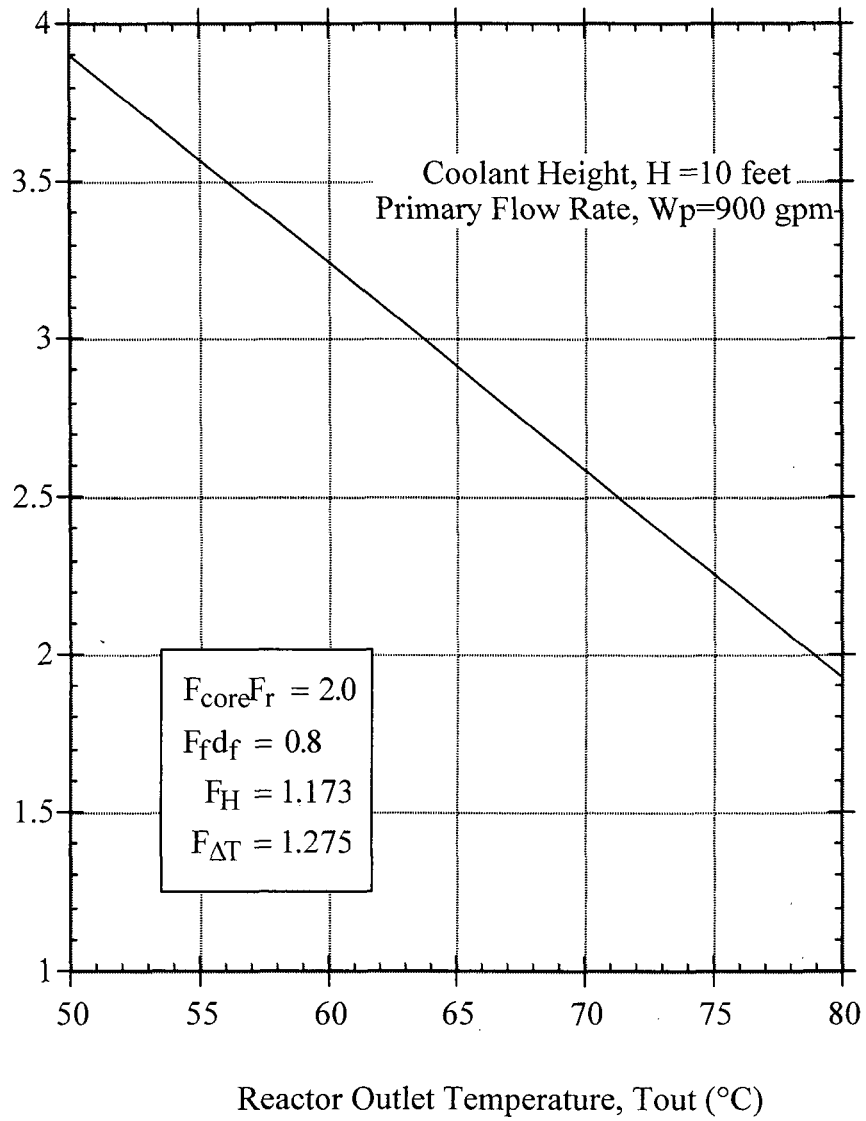


Figure 2.2-2 MITR-II Limiting Safety System Settings for Forced Flow Operation (One Pump).

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.

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.

than 10 $\mu\text{S}/\text{cm}$ 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 $\mu\text{S}/\text{cm}$ 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.

- 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 D₂O 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.

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.

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 $\mu\text{Ci/liter}$ 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

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 are based on experience to date with the MITR-II and on standard practice as recommended in ANSI/ANSI-15.1-2007. Where appropriate, the latter has been modified by the manufacturer's recommendations. Thus, 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.

<u>Monitor</u>	<u>Test</u>
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:

<u>Monitor</u>	<u>Test</u>
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.

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.

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:

- 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

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) ÷ 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, $0.1 \text{ kg U-235} \times (0.993/0.007) = 14.2 \text{ kg U-238}$. Therefore, a limit on the initial amount of U-238 is not required.

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 Director of 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

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.