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TECHNICAL SPECIFICATIONS
FOR THE
MIT RESEARCH REACTOR

MIT-NE-62

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I. INTRODUCTION

The only unique feature of nuclear reactor safety differentiating it from the safety considerations found in any heat producing plant is the fission chain reaction and its radioactive products. The special safety measures taken in nuclear reactor safety result primarily from this unique feature.

The requirements of reactor safety are threefold. First, the general public must be protected. Second, personnel at the reactor must be protected. Third, the reactor plant itself must be protected. The first and second requirements are of primary concern to the AEC as well as to the applicant. The third requirement is of primary concern only to the applicant.

In order to assure the safety of the reactor a defense in depth is set up. This defense in depth is evident in almost every phase of reactor safety. First, every effort is made to prevent an accident. Second, the consequences of an accident are limited and contained by additional safeguards. The defense in depth technique is nicely illustrated by the presence of a minimum of three consecutive enveloping barriers to the release of fission products--the fuel clad, the primary system tanks and piping enclosures, and the outer building containment.

The prevention of a reactor accident can be assured by carefully controlling a number of independent or semi-independent parameters. These parameters control the amount of heat produced per unit time by the fission chain reaction (fission power level), the amount of coolant (D_2O tank level), the flow of coolant to remove the heat (D_2O flow rate), the ultimate disposal of this heat to the secondary system (D_2O outlet temperature), the amount of reactivity that can be added (step reactivity addition), and any possible chemical reaction (D_2 concentration). These six parameters can control the production and removal of heat from the reactor in such a manner as to insure that gross amounts of fission products are not released by melting of the fuel elements under any foreseeable circumstances. Manifestations of all of them are measurable and each one can be controlled by means independent of those used to control the others, except in part the D_2O outlet temperature.

It is conceivable that all of these parameters could have been replaced by a single one--fuel element clad temperature--but that one is not easily and reliably measurable. However, that parameter is the basis of the limits set in those Technical Specifications which follow and which involve prevention of fuel element melting.

Consideration must be given to establishing a consistent set of limiting safe values for the six parameters, henceforth designated herein as the Primary Safety Parameters. For purposes of these Specifications, the limiting safety values will have the meanings shown in Fig. I-1 in which power has arbitrarily been chosen as an example of a Primary Safety Parameter to plot against percent of total reactor operating time spent at each power increment. In MIT's case the licensed reactor power is 5 Mw(t). Therefore, a good operator will endeavor to maintain the reactor as close to 5 Mw(t) as possible. It is almost inevitable that occasionally the operator will exceed the 5 Mw(t) level and this occasional drift over the line is shown by the toe of the curve just

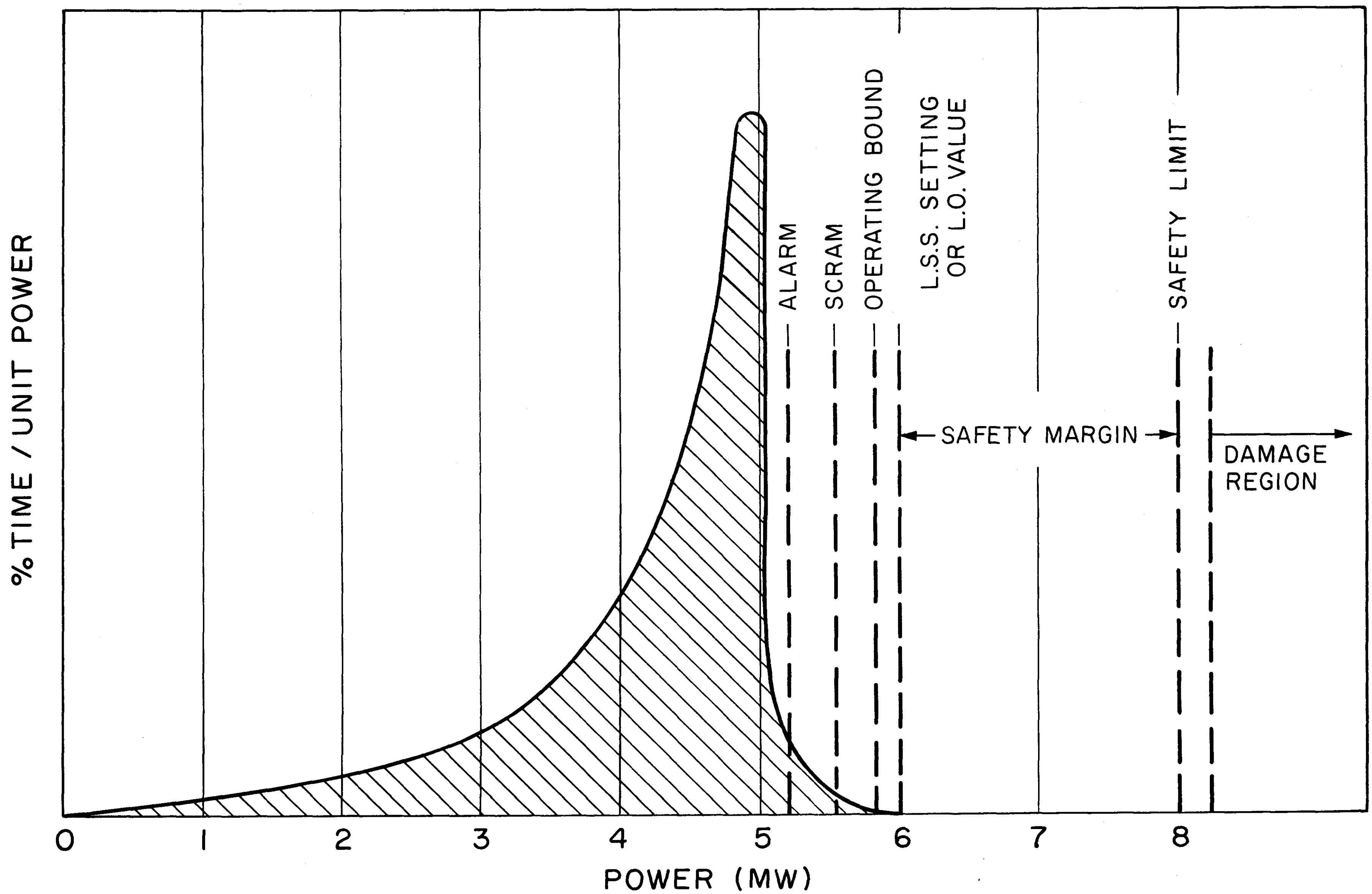
% TIME / UNIT POWER
ALARM
SCRAM
OPERATING BOUND
L.S.S. SETTING
OR L.O. VALUE
SAFETY LIMIT
SAFETY MARGIN
DAMAGE
REGION
0
1
2
3
4
5
6
7
8
POWER (MW)

FIGURE I-1

to the right of 5 Mw(t). This occurrence may also result from slight variations in instrument calibration or for other reasons. At some level, set by MIT, alarms may sound and, at a still higher level, a scram will occur.

Other levels need to be stipulated. One such level is designated as the Limiting Operating Value (LOV). For parameters which are amenable to safety systems and automatic trips, this value will coincide with the Limiting Safety System Setting (LSSS). If this level is exceeded, a report must be prepared and reviewed by the MIT Reactor management, setting forth the circumstances and stipulating any conclusions and corrective action taken. Since this report is available for AEC inspection it seems appropriate that this level constitute the lower bound of the Safety Margin, the upper bound of which is the Safety Limit. The AEC and the applicant are then in agreement as to the extent of the Safety Margin and, further, the AEC will have available to it information of any occurrences in which the Safety Margin is encroached upon.*
As mentioned, the upper bound of the Safety Margin is set by a Safety Limit beyond which it may be unsafe to venture without seriously endangering the system or beyond which there is a marked increase in the probability of an unsafe condition developing. Beyond the Safety Limit is the region where damage to fuel and release of fission products is quite likely.

One further set of parameter values must be stipulated--the values used for setting the reactor conditions to calculate the Safety Limits. These levels should be ones

*The LSSS for most parameters in most reactors will probably be best located above scram levels, but for some parameters in some reactors it may be safer or more convenient to reverse this order.

that are seldom exceeded and then only by small amounts, and they are here designated as the Operating Bounds, and they are obviously the bounds in the direction of non-conservative values. Conceivably, these values could be the nominal or licensed levels, the operating scram points, or the LSS settings. Often these three types of levels will be close together and it will make little difference to the magnitude of the Safety Margin which is chosen. The operating scram point will be a more conservative choice than the licensed level, and the LSS setting still more conservative. In this case, MIT has designated the Operating Bounds for all Primary Safety Parameters as the LSS settings. This is the most conservative choice; but, as indicated above, other choices are possible and may be logically selected with an ample margin of conservatism. The particular choice made by MIT simplifies the Technical Specifications since it combines two different sets of parameter values into one. Thus the alarm and operating scram points remain undesignated in these Specifications, except for the stipulation by MIT that these points will fall within the operating band whose extreme limit is the LSS setting (now defined to coincide with the Operating Bound level).

A maximum plate temperature of 450°C (842°F) has been chosen as the acceptable temperature limit for the calculation of Safety Limits for the other measurable parameters. At this temperature softening of the aluminum beings to be significant, although actual melting does not begin until somewhat above 600°C (1112°F). It is believed that this establishes an adequate Safety Limit well below the point where melting at the hot spot would begin and some fission products would be released. A "real" Safety Limit might be said to exist at the hot spot melting temperature above 600°C.

In order to make a consistent, conservative, and manageable set of Technical Specifications, the systems analysis to determine appropriate Safety Limits is based upon two important assumptions:

(1) Each of the Primary Safety Parameters is independent and controllable by independent means.*

(2) Two independent Primary Safety Parameters will not exceed simultaneously their respective Operating Bounds.

There are two justifications for these assumptions:

(1) By design of the process and safety systems and procedures, two or more simultaneous failures or malfunctions are required to produce significant changes to two Primary Safety Parameters.

(2) The intent of management as evident by the operating procedures is to maintain control of all parameters within the LOV or LSSS.

This means that there must be a misoperation or equipment failure affecting each of two independent parameters* followed by a failure of the monitoring system for each parameter to provide appropriate information and to take corrective action (usually a scram). Thus there must be at least four simultaneous malfunctions or misoperations to invalidate the primary assumptions of these Specifications. The only known important interaction between the Primary Safety Parameters is produced by loss of electrical power which produces an automatic shutdown and, hence, a safer condition.

*The tank outlet D_2O temperature at power is an exception. It was chosen as a Primary Safety Parameter to limit the secondary coolant system conditions, but it also is dependent on the primary system power and flow. For the purpose of this discussion it might almost be considered a secondary parameter monitoring the primary system flow and power and providing additional and independent means of taking correction action for those parameters.

For example, to set a Safety Limit for power, a steady-state reactor was assumed with the flow rate, the reactor D_2O outlet temperature, the D_2O tank level, respectively, just at its Limiting Safety System Setting. Then, in order to establish a Safety Limit on power, it was imagined that all power level safety circuitry was inoperative as power was raised slowly until a calculated power was reached where either the hot spot temperature was calculated to be 450^oC or where conservative predictions indicated an unstable or burnout condition. The lowest power level leading to either of these conditions was chosen as the Safety Limit on power. If doubt existed concerning the validity of information in regard to a certain performance region or calculational method, that method or region was excluded. Thus the analysis was always carried out in a conservative manner.

The situation is similar for other important parameters. For instance, the calculation to set the flow Safety Limit was made with the power, the D_2O tank level, the core D_2O outlet temperature, and the D_2 concentration assumed each to be just at its Operating Bound as the flow was decreased until a plate temperature of 450^oC was reached or until an instability was reached. This flow rate was designated as the Safety Limit. Note that the flow Safety Limit and the others are equally as important as the power Safety Limit, and all must be treated in a similar manner if a scientifically meaningful set of Technical Specifications is to result.

The consideration involving core symmetry was found necessary for the analysis and, therefore, must be a license condition. It is likely that some condition with the same objective will be necessary in other reactors since otherwise asymmetric fuel loadings, small fuel loadings, or asymmetric control rod configurations could lead to localized heat fluxes or heat removal problems much more severe than those envisioned in the calculations. This condition specified in a conservative manner the general nature of the core used in the calculations. In the case of the MITR, this amounted to specifying at powers above 200 kw a core of at least 19 elements controlled by rods banked within 4 in. of one another and fueled with elements of a plate type with a specified acceptable clad thickness, a minimum average void coefficient, and a negative overall temperature coefficient.

These Technical Specifications have been written to incorporate in the "Specifications" themselves and the definitions all requirements which MIT believes are necessary and sufficient to assure the safety of the MIT Reactor. The "Bases" presented with the Specifications set forth the logic and reasoning behind the choice of the Specification in question. They also indicate to the extent possible the uncertainty in the values selected for limits. It is the intent of MIT to present these Bases solely as a backup for the Specification itself--to provide the technical arguments for the Specification chosen. No statement made in any Basis should be construed as a limiting requirement within the license. Once the set of Technical Specifications has been approved, the Bases should not longer be a part of the continuing dialogue between MIT and the AEC unless the logic and reasoning used in the Bases are found by either party to be incorrect and nonconservative.

II. DEFINITIONS

A. Primary Safety Parameters (Reactor)

The following controllable independent variables (except for tank outlet D_2O temperature at power) are designated as "Primary Safety Parameters":

Reactor Primary System Power (neutron flux)

Primary Coolant (D_2O) Volume Flow Rate

Tank Outlet D_2O Temperature at Power (Limits secondary coolant condition, but also depends on power and flow)

Step Reactivity Addition

Tank Level (D_2O) with Reactor Critical

Deuterium Gas Concentration

The several operational conditions of the reactor system mentioned in these Specifications are defined in terms of these variables and Safety Limits are determined in terms of them.

B. Secondary Safety Parameters (Reactor)

The following variables are designated as "Secondary Safety Parameters":

Reactor Period (neutron flux)

Containment External-Internal ΔP

Effluent Air Radiation Levels

Effluent Liquid Waste Radiation Levels

Reactor Room Radiation Levels

Equipment Room Radiation Levels

Exceeding certain operating limits of these Secondary Safety Parameters will result in a scram or other safety action followed by a formal review of the cause of the occurrence. However, no limiting value of these parameters are considered as Safety Limits in the sense defined under F below. Many of these parameters provide information and require safety actions at appropriate levels to insure compliance with AEC regulations 10CFR20 and 10CFR100.

C. Normal Operating Levels

The range of parameter values within which the system normally operates. The normal operating points for all parameters fall within this range and are quite well-designated and closely followed--for instance, the licensed power level is such a point. The limit of the normal operating range in a nonconservative direction for a Primary Safety Parameter is an operating scram level. Operating scram levels must be equally conservative or more conservative than Limiting Safety System Settings, but they are not deemed to be a part of these Technical Specifications and are not contained herein.

D. Operating Bound Limit (OBL)

A bound in the nonconservative direction which sets the other Primary Safety Parameter values to be used as the reactor conditions for calculation of the Safety Limit for one Primary Safety Parameter. (In the MITR these are the same as the LSSS below.)

E. Limiting Safety System Setting (LSSS)

Settings for automatic protective devices related to the Primary Safety Parameters which are so chosen that automatic protective action will correct the most severe abnormal situation anticipated before a Safety Limit is exceeded. If the automatic safety system does not function as required, appropriate corrective action shall be taken and an audit of the abnormal occurrence shall be made by the reactor management. The Commission shall be notified by means of a written report available in the MIT Reactor records, recording the results of the review and corrective action taken and the reasons therefore.

For purposes of these specifications only, the LSSS is defined to coincide with the OBL.

F. <u>Limiting Operating Value (LOV)</u>

Certain parameters, while sufficiently important to be considered Primary Safety Parameters, are not amenable to the concept of System Settings and automatic trips. Protection of the Safety Limits on these variables must be provided by operational control. The Limiting Operating Value is the Primary Safety Parameter value of these certain parameters which, if exceeded, will result in appropriate corrective action and an audit of the abnormal occurrence by the Reactor management. A written report shall be made describing the occurrence, giving the results of the review, and stating any corrective action taken and the reasons therefore.

The LOV is defined to coincide with the LSSS of Primary Safety Parameters which can be protected by automatic safety systems.

G. Safety Limit

A limit set for a given parameter beyond which it may be unsafe to venture without seriously endangering the system or beyond which there is a marked increase in the probability of an unsafe condition developing. The values selected are based on experiments or conservative calculations and are, in general, more conservative than the true or ultimate safety limits. In cases where incomplete information exists these limits are set conservatively. The Safety Limits for the Primary Safety Parameters are set so as to insure that gross fuel melting will not occur within the Safety Limit.

H. In terms of the definitions above, the Primary Safety Parameters shall have the following values:

Primary Safety Parameter	Limiting Safety System Setting	Limiting Value	Safety Limit
Power	6 Mw (2.6 Mw)*	6 Mw (2.6 Mw)*	8 Mw (3.5 Mw)*
Coolant Flow	1800 gpm (750 gpm)*	1800 gpm (750 gpm)*	1350 gpm (600 gpm)*
Tank Outlet Temperature at Power	58°C	58°C	69°C
Tank Level	5 in. below overflow pipe	5 in. below overflow pipe	18 in. below overflow pipe
Conceivable Reactivity Charge	---	2%	2.8%
D_2 Concentration in Helium	---	4%	6%

*Bracketed figures represent levels for 2 Mw operation with only one coolant loop in operation.

H. Normal Reactor Shutdown

The reactor is said to be shut down normally if the following conditions exist:

a) All shim control rods in.

b) Primary coolant system on either normal or shutdown flow.

c) Reactor power at levels set by (γ,n) interactions.

d) Reactor tank level either at overflow level or at dump level.

e) Primary outlet coolant temperature given by $(T) < 53^{\circ}C$.

I. Maximum Security

The reactor facility is said to be in a condition of maximum security if the following conditions exist:

a) All shim control rods in and magnets run down.

b) Ventilation fans off.

c) Quick operating ventilation dampers and back-up dampers closed.

d) Reactor tank D_2O level at or above dump level or fuel under emergency cooling provisions.

e) City make-up water supply to the cooling tower off and lines to City sewer system closed. Cooling tower fans and sprays off with water flowing only into the basin.

f) H_2O City emergency supply checked to assure availability and connected by emergency connection to D_2O system.

g) The containment area and its immediate environs positively secured.

h) Only personnel authorized by the supervisor in charge within the restricted area.

i) In event Specification III.6 is exceeded without consequential effects, it shall be permissible to omit conditions (b) and (c) above.

J. 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.

III. SAFETY LIMITS WITH LIMITING SAFETY SYSTEM SETTINGS OR LIMITING OPERATING VALUES

These conditions assure that fuel plates will not melt and that no other serious harm which could affect the health and safety of operators or the general public will come to the reactor.

III.1. MAXIMUM STEADY STATE POWER LEVEL

Applicability

This specification applies to the steady state thermal reactor power level.

Objective

To ensure that the temperature of the hottest fuel plate will not exceed 450°C (842°F).

Specification

The Safety Limit for the steady state thermal reactor power level shall be 8 Mw. (The Safety Limit for one-loop operation shall be 3.5 Mw.) The Limiting Safety System setting shall be 6 Mw. (The Limiting Safety System Setting for one-loop operation shall be 2.6 Mw.)

Bases

Although aluminum melts at approximately 650°C (1200°F), it begins to soften significantly at about 450°C (842°F) and this temperature is therefore a suitable criterion for guaranteeing the structural integrity of the fuel elements.

Since the MITR operates at atmospheric pressure and since the fuel elements incorporate thin aluminum fuel plates, simple calculations show that the difference of about 350°C between 450°C (842°F) and boiling could not be reached if boiling burnout is prevented.[1] Due to the same reasons and the additional fact that the coolant flow path in the core is of multichannel design, there exists the possibility that flow instabilities could occur before reaching burnout

[1] It is possible that for the conditions present in a core of this type the usual model for predicting burnout ("departure from nucleate boiling" or "film boiling") does not apply and that the heat flux could be pushed even higher before reaching the limiting temperature.

limitations. If flow instability does occur first, it would have the effect of lowering the burnout heat flux to some extent, but the exact amount is at the present time very difficult to predict. In view of this difficulty, it is assumed here that the onset of flow instability causes a lower flow rate in one channel for the same pressure drop and thus causes burnout at a lower heat flux than would be computed from the average flow conditions of the channels. It should be noted that this is a conservative assumption based on the limitations of the present state of the art. This fact combined with the conservative assumptions used in predicting the onset of flow instability produces a safety limit of the maximum steady state thermal reactor power that is considerably below values based on applicable burnout correlations. Thus there appears to be a comfortable margin beyond this safety limit before reaching the point where a real threat to fuel element integrity exists.

The method of investigating the onset of flow instability consists of calculating the pressure drop as a function of flow rate for a given heat flux. The minimum stable flow rate is that which coincides with the minimum of the curve. This is consistent with the stability criterion below:

For instability:

$$\frac{\partial}{\partial W}(\Delta P \text{ external syst.}) - \frac{\partial}{\partial W}(\Delta P \text{ coolant channel}) \geq 0 \quad . \quad (1)$$

Since, for a large number of channels,

$$\Delta P \text{ ext. syst.} = \text{constant} \; ,$$

the criterion reduces to:

$$-\frac{\partial}{\partial W}(\Delta P \text{ coolant channel}) \geqslant 0 \ . \tag{2}$$

This theoretical model has been proposed in several references, (1), (2), and (3), and has recently been found to give excellent agreement with experiment (3).

In the case of the MITR fuel element some interpretation is necessary in order to define the coolant channel. The entire fuel element could be considered a coolant channel, since there are 19 to 30 elements in the core and the assumption of a constant pressure drop across the fuel elements is not significantly affected by flow changes in any one element. The alternative is to treat the individual channels between fuel plates of which there are 17 per standard element, and less for reduced plate elements. This distinction has considerable bearing on the results and as will be seen below the latter case is the more conservative and has therefore been used in this analysis.

The pressure drop across the coolant channel can be divided into four parts:

(1) the entrance effect

(2) the friction factor drop along the length of the channel

(3) the exit effect

(4) the gravity drop--since the flow is vertical and upward.

In the absence of boiling, these pressure drops are normally computed by standard methods and, except for the last one which is independent of flow rate, are an increasing function of flow rate, i.e., the slope of the pressure drop vs. flow curve is positive and there is no flow instability. The presence of boiling in the channel affects all except possibly the first factor. In general, the entrance effect and the exit effect are increasing functions of flow rate, but their magnitudes may be decreased by the transition to two-phase flow. The friction drop may have either a positive or negative slope and may be considered the determining factor.

Having defined the coolant channels as the individual channels between fuel plates, the entrance effect becomes both small and difficult to calculate due to the geometry of the MITR fuel element lower adapter. For these reasons this pressure drop has been neglected, introducing a conservative factor.[2] Since this term would be relatively large and have a positive slope with respect to flow rate for the case of an entire element considered as a coolant channel, the selection of the individual fuel plate channels as the coolant channels is quite conservative.

The exit effect pressure drop is also negligible for the individual channels, but would be significant for the whole element. It has therefore also been neglected and, for reasons similar to those given above, this represents another conservative factor.

Similarly, the gravity drop becomes an increasing function of flow rate for two-phase flow, but requires knowledge of the void fraction which is extremely difficult to predict. Neglecting this term then adds still another conservative factor.

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[2] A relatively simple fuel element modification incorporating orificing for the coolant channels could probably produce a substantial improvement in stability by adding an entrance effect pressure drop.

The remaining term is the friction term, which for the no-boiling case is normally computed by:

$$\Delta P_{ADB} = f \frac{L}{D} \frac{V^2}{2g} \quad , \qquad (3)$$

where: f = dimensionless friction factor;

L/D = dimensionless length to diameter ratio = 112;

V = velocity, ft./sec.;

g = conversion factor, 32.3 ft.-lb. force/lb. mass sec.^{2}; and

ΔP_{ADB} = pressure drop, ft. of coolant.

The friction factor f has recently been measured for an MITR fuel element by L. R. Enstice (4), and over the flow range of interest found to be a constant = 10^{-2}. This pressure drop is more complicated for the boiling case, however, and must be modified accordingly. Dormer (5) has experimentally measured this effect for small round tubes for subcooled boiling over a wide range of parameters and found that the data can be best correlated by plotting:

$$\Delta P/\Delta P_{ADB} \text{ vs. } q/q_{sat} \quad ,$$

where: ΔP, ΔP_{ADB} = pressure drops with and without boiling, respectively;

$\frac{\Delta P}{\Delta P_{ADB}}$ = correction factor to be applied to the results of Eq. (3);

q = heat on surface of coolant channel;

q_{sat} = heat that would produce saturated boiling at the exit of the channel.

Dormer's results are plotted for various L/D ratios in Fig. III.1-1. It can be seen from the data in Appendix A that Dormer's data involve parameters that are reasonably consistent with the case of interest. Probably the most questionable factor involving its use in the present case is that the data were obtained for round tubes, while the coolant channels of interest are narrow rectangular channels. Griffith has stated (6), however, that the concept of a hydraulic diameter is meaningful for this purpose, so that the use of the data is justified.

If all other parameters are unchanged, a reduction in flow will decrease q_{sat} and thereby increase q/q_{sat}, hereafter referred to as R. Referring to Fig. III.1-1, if R is sufficiently large (for the prescribed L/D curve), $\Delta P/\Delta P_{ADB}$ is an increasing function of R. This behavior may be interpreted as a pressure drop vs. flow relationship with a negative slope as opposed to the positive slope of Eq. (3). The flow rate at which the product of ΔP_{ADB} and $\Delta P/\Delta P_{ADB}$ is a minimum then corresponds to the minimum stable flow rate for the prescribed conditions as, for example, in Fig. III.1-2.

The actual calculation of R used to enter Fig. III.1-1 involves other system parameters, specifically: reactor power, the power produced in the hottest plate of the hottest fuel element, the D_2O flow rate, and the D_2O inlet temperature. The D_2O flow rate has been fixed at 1800 GPM, the OBL value. The power produced in the hottest plate of the hottest fuel element is obviously a function of power and has been taken to be also a function of the number of elements in the core. This is meaningful since for a given reactor power the power per plate is decreased if the number of fuel elements is increased. To be consistent, this relationship has been taken to be equivalent to that established for the prevention of nucleate boiling at 6 Mw with a D_2O outlet temperature of 55^oC (7) then:

$$P_{PMAX} = \frac{500(1.25)}{(16)} \left(\frac{19}{N}\right)^{0.8} \frac{P_T}{5000} , \qquad (4)$$

$$P_{PMAX} = 7.82 \times 10^{-3} \left(\frac{19}{N}\right)^{0.8} P_T , \qquad (4a)$$

where: P_{PMAX} = maximum power per hottest plate (Mw),

N = number of elements in the core, and

P_T = reactor thermal power (Mw).

This relationship is counterbalanced by the effect of the decreased flow per coolant channel when the number of fuel elements is increased and the total D_2O flow rate is kept constant at 1800 GPM. The net effect of the dependence upon the number of elements is to make the 30-element core (maximum number) the closest to instability and the final calculations are based on this case. It should also be noted that the use of Eq. (4) establishes hottest plate power values which are considerably higher than those predicted by Devoto (8), including hot channel factors, and are therefore conservative in that sense.

Since reactor power is a variable parameter for the purpose of this calculation, there remains only the relationship with the D_2O inlet temperature. At a given time the actual value of the D_2O inlet temperature will be determined by a number of factors, such as the outside wet bulb temperature, the cooling tower effectiveness, the H_2O flow rate, the overall heat exchanger heat transfer coefficient, and the D_2O flow rate. For a given power and flow, however, there is a simple relationship between the inlet and outlet temperature, and since Eq. (4) gives the hottest plate power as a function of reactor power, the ratio R can be determined independent of the H_2O system provided a limiting D_2O outlet temperature is established. This limiting temperature for this calculation is 58°C, the OBL value for the D_2O outlet temperature.

This leads to the following derivation for R:

$$P_T = W_T C_P (T_{out} - T_{in})$$

$$= (2.915 \times 10^{-4}) W_T (T_{out} - T_{in}) \ , \qquad (5)$$

where: P_T = reactor thermal power, Mw;

W_T = total D_2O flow rate, GPM; and

T_{out}, T_{in} = D_2O temperature, °C.

For W_T = 1800 GPM and temperatures in °F,

$$P_T = 0.2914 (T_{out} - T_{in}) \ . \qquad (5a)$$

Now,

$$P_{sat} = W_c C_p (T_{sat} - T_{in}) \ , \qquad (6)$$

where: P_{sat} = power required to raise the bulk coolant temperature in the hottest channel to saturation,

W_c = D_2O flow rate in the hottest channel, and

T_{sat} = saturation temperature at channel exit.

The flow through an individual channel W_c is related to the total flow, W_T, through the number of elements in the core:

$$W_c = \frac{W_T}{17N} \ . \qquad (7)$$

Then, by analogy with Eq. (5a),

$$P_{sat} = \frac{0.2914}{17N} (T_{sat} - T_{in})$$

$$= \frac{1.714 \times 10^{-2}}{N} (T_{sat} - T_{in}) \ . \qquad (6a)$$

Using Eq. (5a) to substitute for T_{in} gives:

$$P_{sat} = \frac{1.714 \times 10^{-2}}{N} (T_{sat} + 3.42\ P_T - T_{out}) \ . \qquad (6b)$$

Now, by definition,

$$R = \frac{q}{q_{sat}} = \frac{P_{PMAX}}{P_{sat}} , \qquad (8)$$

$$= \frac{7.82 \times 10^{-3} \left(\frac{19}{N}\right)^{0.8} P_T}{\frac{1.714 \times 10^{-2}}{N} (T_{sat} + 3.43\ P_T - T_{out})} , \qquad (8a)$$

$$= \frac{4.81\ N^{0.2} P_T}{(T_{sat} + 3.43\ P_T - T_{out}} \ . \qquad (8b)$$

Since the H_2O saturation temperature is approximately 215°F for a pressure of 15.5 psia (local pressure at channel exit), this is a conservative value for D_2O.

Taking T_{sat} = 215°F, and N = 30,

$$R = \frac{9.494\ P_T}{(215 + 3.43\ P_T - T_{out})} \qquad (8c)$$

Eq. (8c) is the desired relationship.

It remains now to determine the value of R which corresponds to a minimum stable flow condition. For constant power and outlet temperature, R is inversely proportional to flow and thus, from Eq. (3), ΔP_{ADB} varies as $1/R^2$. The pressure drop with boiling, ΔP, is then proportional to the quantity $(\Delta P/\Delta P_{ABD})\ 1/R^2$ and the minimum value of this quantity will coincide with conditions of minimum stable flow. Using the data of Fig. III.1-1, Fig. III.1-2 was constructed which indicates that a minimum stable flow condition occurs with a value of R of about 0.72.

Using this value of R and Eq. (8c), the plot in Fig. III.1-3 is obtained. This indicates that stable conditions exist for a total flow of 1800 GPM provided operating conditions correspond to a point below and to the left of the line of Fig. III.1-3. As can be seen from the graph, the Safety Limit of 8 Mw is consistent with the OBL value of 58°C for the D_2O outlet temperature.

The conditions at 8 Mw must now be checked to ensure that the flow instability does in fact occur prior to burnout. References (9) and (10) are among the most recent burnout correlations which completely bracket the physical parameter values for the MITR fuel element channels. Calculations are given in Appendix A for the 8 Mw core which is summarized below:

Power - 8 Mw

No. Fuel Elements - 30

Total D_2O Flow - 1800 GPM

Flow/Element - 60 GPM

Flow/Coolant Channel - 3.53 GPM

Maximum D_2O Outlet Temperature - 58°C (136°F)

Maximum D_2O Inlet Temperature - 43°C (109°F)

Power Produced in Hottest Plate - 43.4 kw

Average Heat Flux in Hottest Plate -
1.70×10^5 Btu/hr. ft.^{2}

Axial Flux Ratio, Avg./Max. - 0.8118 (Ref. (11))

Maximum Heat Flux in Hottest Plate -
2.90×10^5 Btu/hr. ft.^{2}

The results of the burnout correlation calculations are given below:

Correlation	Burnout Heat Flux (Btu/hr. ft.^{2})
Gambill (9)	1.41×10^6
Macbeth (10)	1.6×10^6

The two correlations are in good agreement and while it is not clear whether the results should be compared with the average or the maximum heat flux in the hottest plate, even the maximum value is exceeded by a factor of about 5. It can be concluded, therefore, that boiling burnout will not occur for the 8 Mw core considered.

The Limiting Safety System (LSS) setting of 6 Mw is based on calculations of that power which will just allow the plate surface temperature at the hot spot to reach 100^{o}C. In this way, a very conservative limit has been established that precludes any possibility of boiling in the core. It should be noted here that no credit has been taken for the film temperature drop. In addition, nucleate boiling in the core should have no adverse effect and, in fact, it is reasonably certain that the reactor could be raised to power well above 8 Mw without encountering any serious adverse effects.

In addition, the 6 Mw level is far enough above the 5 Mw licensed power to permit the setting of power scram levels at intermediate values between 5 Mw and 6 Mw, and yet 6 Mw is not so far away from 5 Mw as to significantly affect the hypothetical accident conditions and fission product releases calculated.

It should be pointed out that this choice of 6 Mw for the LSSS and 8 Mw for the Safety Limit is somewhat arbitrary and could be raised, for instance, by cutting into the flexibility allowed here for future fuel element design. The choice of any parameter Safety Limit or LSSS in the case of flow, power, and T_{out} depends on its interaction with the other two. In particular, the choice of the 8 Mw Safety Limit depends on the acceptable (LSSS) coolant outlet temperature, the acceptable (LSSS) coolant flow, and the maximum acceptable power developed per fuel element plate. The choices made of the limits are designed to correspond to the present well-verified operating conditions while still providing reasonably large margins between the Safety Limits and the LSSS levels and providing sufficient flexibility in core loading patterns and fuel designs.

APPENDIX A - Sample Calculations of Burnout Correlations

Gambill (9) gives the following correlation for the burnout heat flux:

$$\phi_{Bo} = KL_v\rho_v \left(\frac{\sigma g_c a \Delta\rho}{\rho_v^2}\right)^{1/4} \left[1 + \left(\frac{\rho_\ell}{\rho_v}\right)^{0.923} \left(\frac{C_p \Delta T_{sub}}{25 L_v}\right)\right]$$

$$+ K'\left(\frac{k}{D}\right) N_{Re}^m N_{Pr}^n (t_w - t_b)_{Bo} , \qquad \text{(A1)}$$

where: K, K' = adjustable constants of boiling and convective terms;

L_v = latent heat of vaporization;

ρ_v = density of vapor;

σ = surface tension;

g_c = conversion constant, $L \cdot M/F \cdot \theta^2$;

a = local acceleration;

$\Delta\rho$ = density difference $(\rho_\ell - \rho_v)$;

C_p = constant pressure specific heat of liquid;

ρ_{ℓ} = density of liquid;

ΔT_{sub} = degree of subcooling $(t_{sat}-t_b)$;

k = thermal conductivity of liquid;

D = equivalent diameter of flow passage;

N_{Re} = Reynold's number;

N_{Pr} = Prandtl number;

m,n = exponents; and

t_w = wall temperature at burnout.

This correlation is based on a physical model and the first term corresponds to the boiling heat transfer at burnout while the second term corresponds to non-boiling or convective heat transfer at burnout. The use of the second term requires knowledge of t_w, the wall temperature at burnout, and the method proposed for predicting (12) t_w does not cover the temperatures of interest for low pressure cases. This apparently is a result of the fact that the convective term is significant only for the higher pressures. In any event, this term has been neglected in the present case, and any error so introduced will be conservative.

Table III.1-3 below gives the extreme ranges of some of the physical parameters covered by the data correlated.

Table III.1-3 Range of Date (for rectangular channels) for Ref. (9)

Parameter	Range
Pressure (psia)	14-2,000
ΔT_{sub} (^{o}F)	0-282
v (ft./sec.)	4.8-85.4
ϕ_{Bo} (Btu/hr.ft.^{2}x10^{-6})	0.42-11.41

Table III.1-4 below gives the values of the parameters used in Eq. (A1).

Table III.1-4 Parameters for Eq. (A1) for 8 Mw Core

$K = 0.14$

$\sigma = 58.85$ dynes/cm. $= 4.032 \times 10^{-3}$ lb./ft.

$a = g_c = 32.2$ ft./sec.^{2} $= 4.17 \times 10^{8}$ ft./hr.^{2}

$L_v = 890$ Btu/lb.

$\rho_\ell = 65.71$ lb./ft.^{3}

$\Delta\rho = 65.67$ lb./ft.^{3}

$\rho_v = 4.04 \times 10^{-2}$ lb./ft.^{3}

$C_p = 1.0$ Btu/lb.^{o}F

$D_e = 0.2124$ in.

$\Delta T_{sub} = 67.8^{o}F$ (at hot spot of hottest channel)

The results of Eq. (A1) give a burnout heat flux of 1.41×10^{6} Btu/hr.ft.^{2}.

Macbeth (10) has correlated a wide range of burnout data using slightly different equations for round tubes and rectangular channels.

For round tubes:

$$\phi \times 10^{-6} = \frac{A + \frac{1}{4}\, CD\, (G \times 10^{-6})\, \Delta h_i}{1 + CL}\,, \tag{A2}$$

where:

$$A = y_0 D^{y_1} (G \times 10^{-6})^{y_2}\,;$$

$$C = y_3 D^{y_4} (G \times 10^{-6})^{y_5}\,;$$

ϕ = burnout heat flux, Btu/hr.ft.^{2};

D = tube diameter, in.;

G = average mass velocity, lb./hr.ft.^{2};

Δh_i = subcooled enthalpy at channel inlet, Btu/lb.;

L = channel length, in.; and

$y_0 \longrightarrow y_5$ = optimized parameters (functions of pressure, flow, and geometry).

For rectangular channels:

$$\phi \times 10^{-6} = \frac{A + 0.555\, CS\, (G \times 10^{-6})\, \Delta h_i}{1 + CL}\,, \tag{A3}$$

where:

$$A = y_0 S^{y_1} (G \times 10^{-6})^{y_2}\,,$$

$$C = y_3 S^{y_4} (G \times 10^{-6})^{y_5}\,, \text{ and}$$

S = internal spacing between flat heating surface rectangular channel, in.

Table III.1-5 below gives the ranges of data correlated.

Table III.1-5 Range of Data[3] Correlated in Ref. (11)

Channel	Parameter	Range
Round tubes	Pressure (psia)	15-2750
	L (in.)	1-79
	D (in.)	0.40-0.940

[3]For both tubes and rectangular channels, the "high velocity regime" is the appropriate category. The "low velocity regime" data is therefore not considered here.

Table III.1-5 (continued)

Channel	Parameter	Range
Round tubes	$G \times 10^{-6}$ (lb./hr.ft.^{2})	0.11-7.82
	Δh_i (Btu/lb.)	2-713
Rectangular channels	Pressure (psia)	600-2000
	L (in.)	6-27
	S (in.)	0.050-0.101
	$G \times 10^{-6}$ (lb./hr.ft.^{2})	0.1-4.78
	Δh_i (BTU/lb)	12-654

The minimum pressure used in the rectangular channel data is 600 psia, and this raises some doubt as to the applicability of the correlation for the core considered here. However, Rohsenow (13) has proposed the following procedure for use with no significant loss of accuracy:

(1) The round tube correlation is used for pressures of 15 psia and 530 psia, using the appropriate Δh_i values.

(2) A pressure correction factor, F_p, is obtained from Step (1) above by dividing the 15 psia value by the 530 psia value.

(3) The rectangular channel correlation is used with a pressure of 600 psia and then multiplied by F_p.

The values used in Eqs. (A2) and (A3) are given in Table III.1-6:

Table III.1-6 Parameters for Eqs. (A2) and (A3) for 8 Mw Core

$D = 0.2124$ in.

$G = 0.964 \times 10^6$ lb./hr.ft.^{2}

$L = 24$ in.

$S = 0.111$ in.

$\Delta h_i = 120, 380,$ and 393 Btu/lb. (for 3 cases calculated)

Table III.1-7 gives the results of the three cases calculated.

Table III.1-7 Results of Eqs. (A2) and (A3) for 8 Mw Core

	Round-15 psia	Round-530 psia	Rectangular-600 psia
y_0	1.12	1.57	23.5
y_1	-0.211	-0.566	-0.472
y_2	0.324	-0.329	-3.29
y_3	0.001	0.0127	0.123
y_4	-1.4	-1.4	-1.4
y_5	-1.05	-0.737	-3.93
A	1.5339	3.82	74.47
C	0.009033	0.1145	3.0904
$\phi \times 10^{-6}$	1.307	1.613	1.976

The pressure correction factor is then:

$$F_p = \frac{\phi \text{ 15 psia}}{\phi \text{ 530 psia}} = \frac{1.307 \times 10^6}{1.613 \times 10^6} = 0.8102 \text{ ,}$$

and the corrected burnout heat flux is then:

$$\phi = \phi \text{ 600 psia} \cdot F_p = (1.976 \times 10^6)(0.8102)$$
$$= 1.60 \times 10^6 \text{ Btu/hr.ft.}^2 \text{ .}$$

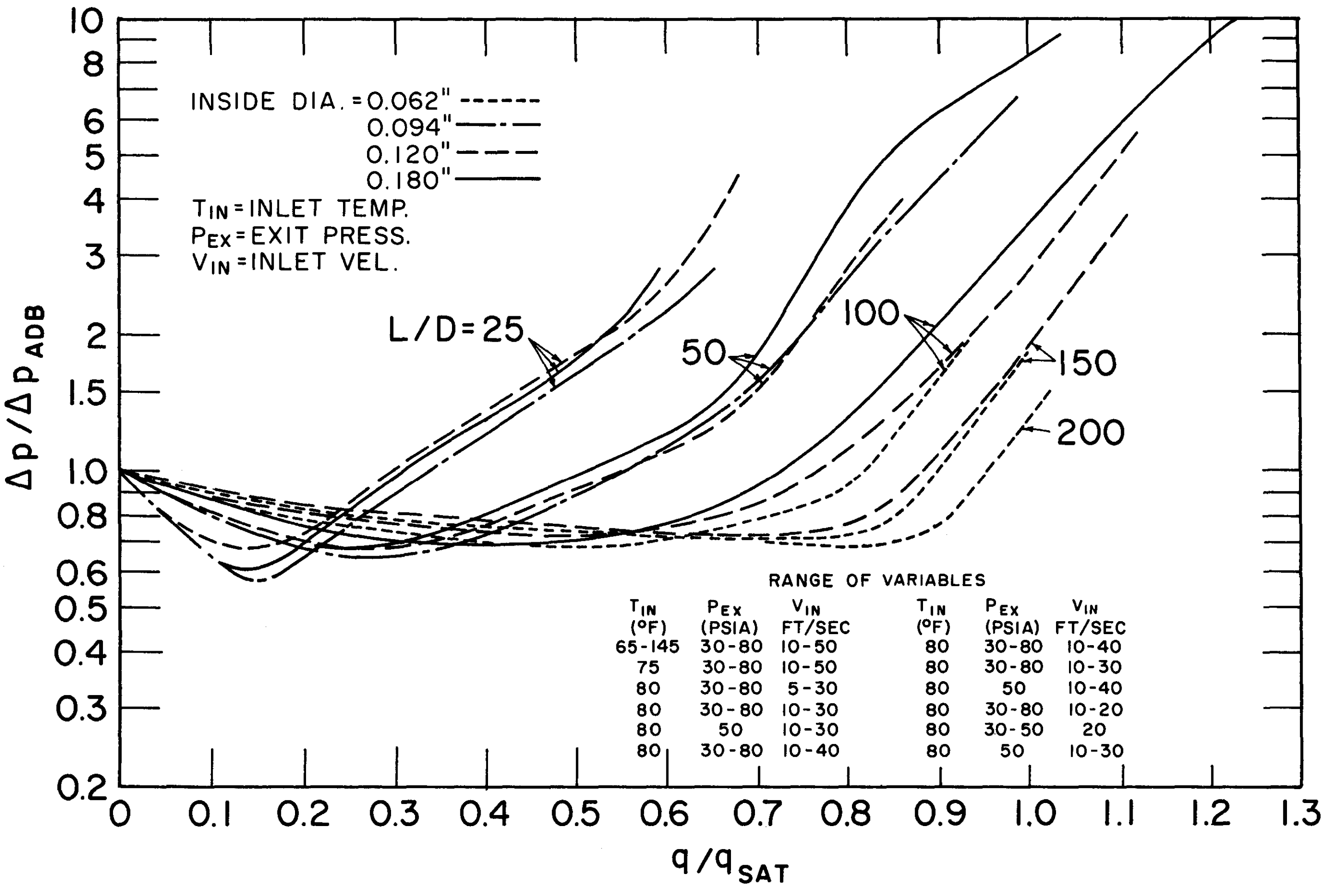

FIGURE III. 1-1 CORRELATED PRESSURE DROP - ALL GEOMETRIES (REF. (5))

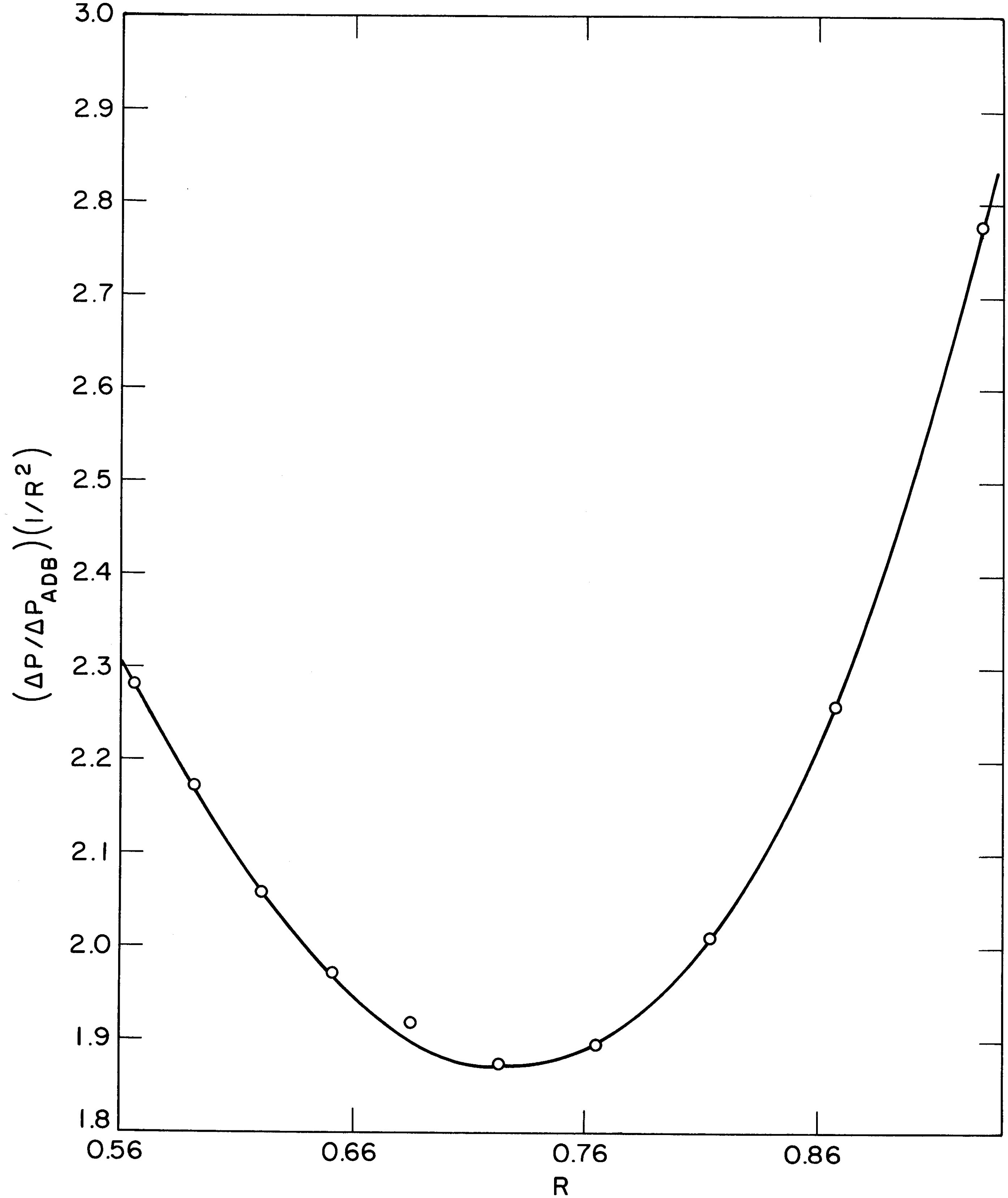

FIGURE III.1-2 DETERMINATION OF MINIMUM STABLE CONDITION

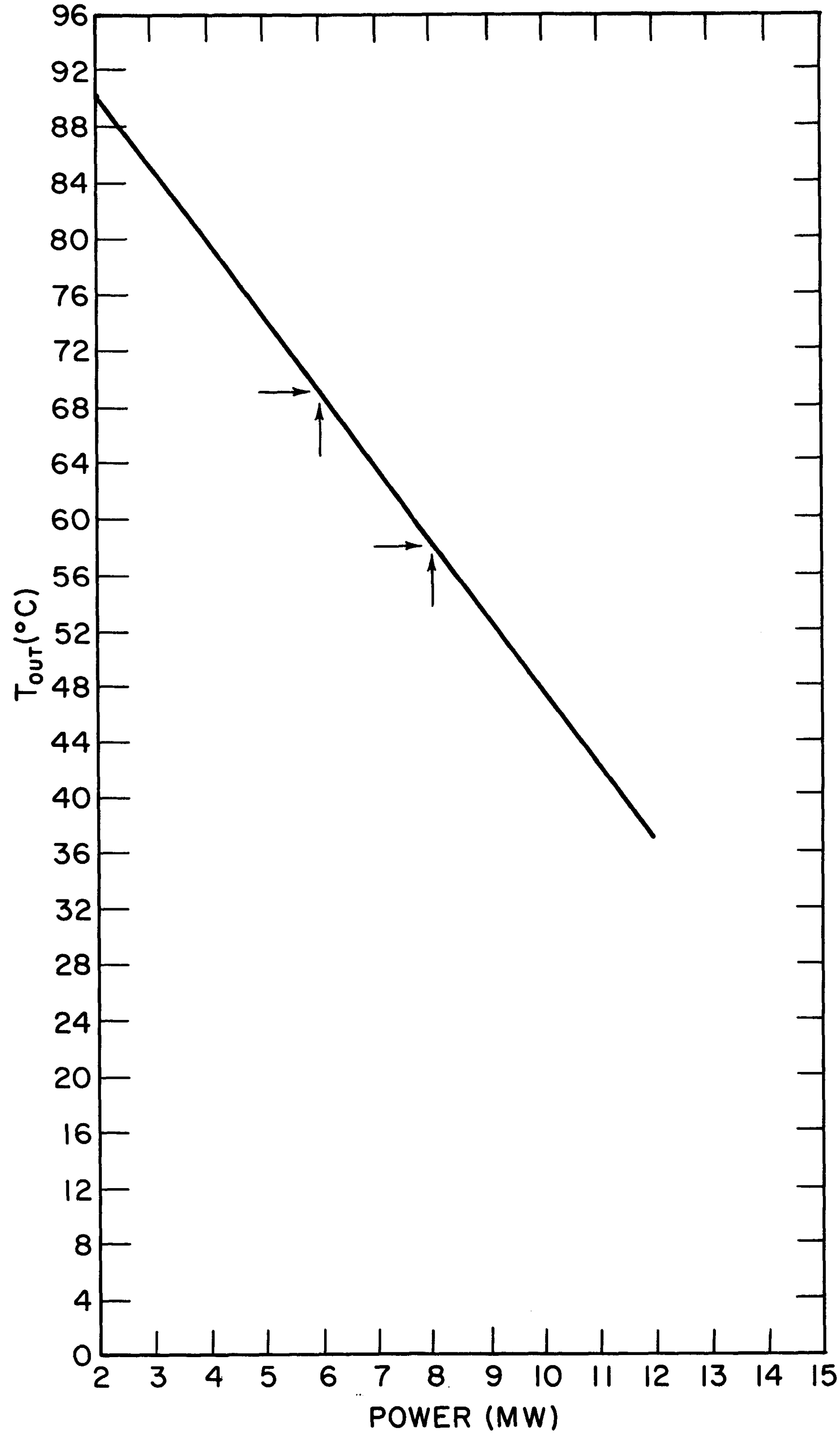

FIGURE III.1-3 MAX. D_2O OUTLET TEMPERATURE VS REACTOR POWER FOR STABLE FLOW CONDITIONS AT 1800 G.P.M. TOTAL FLOW

III.2 MINIMUM D_2O FLOW RATE

Applicability

This specification applies to the D_2O flow rate.

Objective

To ensure that the temperature of the hottest fuel plate will not exceed 450°C (842°F).

Specification

The Safety Limit for the D_2O flow rate shall be 1350 GPM through the fuel elements. (The Safety Limit for one loop operation shall be 600 GPM.) The Limiting Safety System setting shall be 1800 gpm. (The Limiting Safety System setting for one loop operation shall be 750 gpm.) This specification does not apply to operation below 200 Kw.

Bases

Using the Operating Bound values for reactor power and D_2O outlet temperature, 6 Mw and 58°C, the minimum stable D_2O flow is 1350 GPM, using the same approach as in Section III.1. (In a similar manner, 2.6 Mw and 58°C gives a minimum stable flow of 600 GPM.)

The Limiting Safety System setting of 1800 GPM for coolant flow has been established, first, on the basis that flows in excess of this value can be attained easily in normal operation. Second, at flows less than this value and with a tank outlet temperature of 58°C there is a possibility of the onset of flow instability at power levels greater than 8 Mw. As indicated in Section III.1 the selection of OB levels and Safety Limits on power, flow, and outlet D_2O temperature are interrelated.

At power levels below 200 Kw, coolant flow is not necessary. (See Bases of Section IV.3).

III.3. MAXIMUM D_2O OUTLET TEMPERATURE

Applicability

This specification applies to the D_2O outlet temperature.

Objective

To ensure that the temperature of the hottest fuel plate will not exceed 450°C (842°F).

Specification

The reactor shall not be critical with the D_2O outlet temperature greater than 69°C. The Limiting Safety System setting shall be 58°C.

Bases

By using the Operating Bound levels for reactor power and D_2O flow rate, 6 Mw and 1800 GPM, the minimum stable flow condition is found when the D_2O outlet temperature is raised to 69°C. The calculations leading to these values utilize the methods of Section III.1. The results are shown in Fig. III.1-3 of Section III.1.

The Limiting Safety System setting of 58°C for the D_2O core outlet temperature has been established, first, on the basis that values of the outlet temperatures of 35-50°C depending upon cooling tower conditions are those normally observed in the operation of this system and, hence, 58°C would be sufficiently above normal to make an audit of the situation fruitful. Second, the choice of this value permits the establishment of satisfactory limits for power, coolant flow, and fuel element design.

III.4. MAXIMUM SAFE STEP REACTIVITY ADDITION

Applicability

This specification applies to step reactivity additions.

Objective

To ensure that the surface temperature of the hottest fuel plate will not exceed 450°C during any credible reactivity excursion.

Specification

The maximum amount of reactivity that may be added in a stepwise manner 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 in such a manner shall not exceed 2.8% in reactivity. The Limiting Operating Value shall be 2%.

Bases

Technical Bases

The present "Maximum Credible Accident" to the MITR is defined in the 5 MW Report (7) as the sudden insertion of a 160-gram element into the central position of a just critical core which would add 2.8% reactivity.

The subsequent formation of steam voids in the core will shut the reactor down before melting occurs in the fuel elements. The maximum fuel plate temperature was estimated to be about 300°C by using the approach outlined in Ref. (14), based on experiments conducted on H_2O-moderated cores in the Borax and Spert programs.

Since that report, transient experiments have been conducted on D_2O-moderated cores as part of the Spert program (15,16,17). Two D_2O cores differing in neutron lifetime and void coefficient were investigated. In both cases the fuel element design and enrichment were quite similar to that of the MITR element with the Spert element having more closely spaced fuel plates. The pertinent nuclear properties for the cores are given below.

	SPERT		MITR
	Close Packed	Expanded	
ℓ/β	0.064 sec.	0.10 sec.	0.17 sec. (Ref. (18))
C_v	255 mβ/liter	200 mβ/liter	214 mβ/liter (Ref. (19))

where: ℓ = neutron lifetime,

β = delayed neutron fraction including photoneutrons (0.75% ρ in MITR), and

C_v = average void coefficient.

It was found that the two Spert cores exhibited the same transient behavior for a given period as seen in Fig. III.4-1 (16) over the range investigated with no coolant flow. On extension of the expanded core measurements to shorter periods the experiment shows a rapid rise in maximum temperature at a reciprocal period of about 18 $sec.^{-1}$ which is assumed to be due to departure from nucleate boiling (DNB) at the surface of the plates. Unfortunately, the departure from nucleate boiling (DNB) was not observed in the close packed core during the "no flow tests" because they were not extended to sufficiently short periods. It is concluded that the curve shown in Fig. III.4-1 will apply to the MITR up to the DNB point since the Spert cores were found to agree over a range of lifetimes and void coefficients.

Some estimate of the behavior of the MITR past the DNB point may be made by considering the following Spert data:

(1) In an H_2O-moderated core with $\ell/\beta = 8.16 \times 10^{-3}$ sec., melting was observed with a 5-msec. period (20).

(2) From the curve shown in Fig. III.4-1, melting in the expanded D_2O core may be estimated to occur at about a 30-msec. transient.

Since the neutron lifetime for the MITR is longer than either of these cores, it is expected that melting would occur at a somewhat longer period. A linear extrapolation of period versus ℓ/β predicts melting of the MITR plates at about a 50-msec. period. By extrapolation of the expanded core data in Fig. III.4-1 a plate temperature of 450°C would be reached at $\alpha = 28$ sec^{-1} and if the reciprocal period is increased by 4 sec^{-1} the melting temperature (660°C) is estimated to occur. Assuming a similar relationship for the MITR, with a 20 sec^{-1} predicted reciprocal period for melting to occur, then 450°C would be reached when $\alpha = 16$ sec^{-1} which corresponds to about a 62.5-msec. period and an insertion of 3.7 β or 2.8% $\delta k/k$ in reactivity.

During the Spert excursions on the expanded core, the pressure at time of maximum power and the maximum pressure attained sometime later were measured. With $\alpha = 16.1$ sec^{-1}, (which could be obtained with a 2.8% insertion into the MITR) the Spert reactor produced pressures of 5.5 and 24 psig, respectively. These pressures are well within the MITR design pressure of 40 psig.

It is then concluded that even a 2.8% reactivity insertion into the MITR with no flow would not endanger the integrity of the fuel elements.

The effect of flow on the temperatures attained is shown in Fig. III.4-2 (17) for experiments conducted on the close packed core. No flow tests were conducted on the expanded core. It is seen that the maximum temperature attained is about the same for both flow and non-flow cases, but the post-peak behavior is significantly different. The average power level and frequency of oscillations increase with increasing flow. This effect led to fuel element melting about 3 sec. after initiation of the period at rather low reactivity insertions in the Spert compact core.

The MITR is protected from such instabilities by automatic instrument scrams actuated by period and neutron power levels. The period scram would occur very shortly after the initiation of the transient while the level scrams would actuate when the neutron level had reached the scram level of 6 Mw. From Fig. III.4-2 it is seen that a response time of 1.5 sec. (as specified in Specification V.4) would be sufficient to prevent secondary oscillations.

The Limiting Operating Value of 2% in reactivity is based on establishing an adequate Safety Margin below the Safety Limit of 2.8%. In particular, a study of the Spert II data would seem to indicate that a transient involving the stepwise addition of 2% in reactivity could be accommodated safely under flow conditions of 6 to 7 ft./sec. including the flow-power instability oscillation effect observed at Spert II following the initial transient. Thus, on the basis of the Spert tests, even in case the reactor were to fail to scram within 1.5 sec. as it should, the transient and the subsequent oscillation should not result in core damage.

The safety margin of 0.8% Δk is sufficient to accommodate even the total withdrawal of the regulating rod within this margin. The rod is the only one utilizing automatic control. While it should be thrown out of automatic control by the electronic circuitry, it is conceivable that it might not. The maximum reactivity controllable by the regulating rod has been set at 0.75% (equal to β in this reactor). Normally its worth is approximately 0.6%. It does not appear credible that there would be any interaction between a change in reactivity caused by an experiment and the operation of this regulating rod so as to cause it to drive out. (The drive is a rack and pinion system with a withdrawal rate of less than 1.0 in./sec. corresponding to a reactivity effect of 2×10^{-4} $\delta k/k$/sec.) In addition, the regulating rod is normally operated in its mid range. The slow rate of withdrawal, the total worth of the regulating rod, and the incredibility of an interaction all indicate the conservatism of the margin from the viewpoint of the operation of the regulating rod.

Assessment of Calculations

The calculations of the plate temperature attained on step reactivity insertions is limited primarily by predicition of behavior above the DNB point. The method outlined above is consistent with the few cases where this phenomenon was observed. Future measurements and/or theoretical interpretation may improve the technical bases of this specification.

The effect of void coefficient on the maximum temperature appears to be quite weak. The two D_2O cores studied at Spert which differed somewhat in void coefficient exhibited almost identical behavior. Using a correlation of H_2O results (20) (which is applicable to D_2O cores if experimental pressures are used), the effect of decreasing the void coefficient to 150 mβ/liter increases the predicted temperature at 2.8 $\delta k/k$ by about 25°C if no steam blanketing occurs. It is then concluded that a variation of 25% in the MITR void coefficient would not affect this specification. Therefore, 170 mβ/liter is chosen as a conservative lower limit for void coefficient.

In the expanded D_2O core with $\ell/\beta = 0.1$, a temperature of 450°C on the plates would be reached with $\alpha = 28$ sec^{-1} which corresponds to an insertion of 3.8 β or 2.9% $\delta k/k$. The permissible reactivity insertion therefore does not appear to vary significantly with variations in neutron lifetimes typical of these D_2O cores.

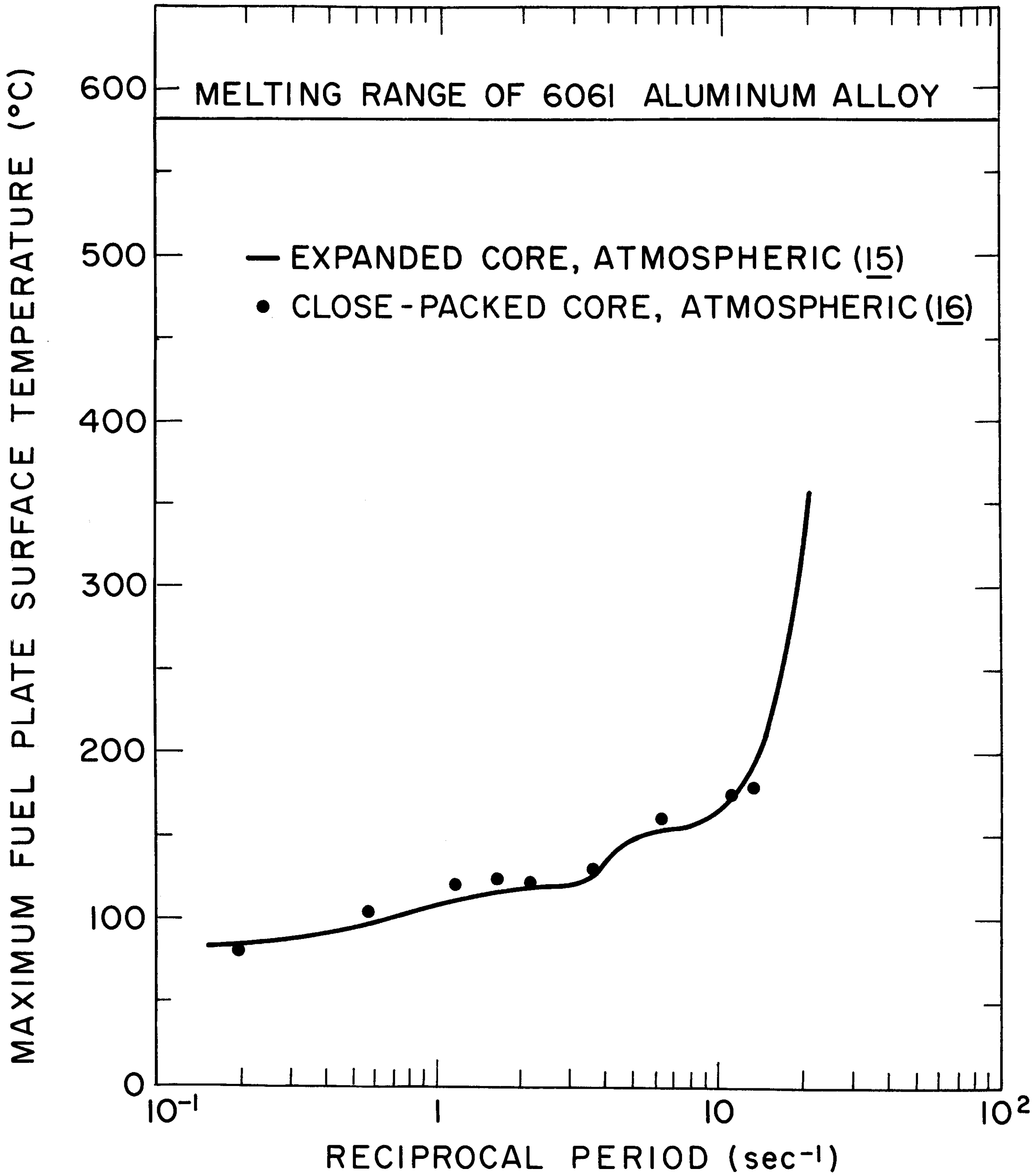

FIGURE III.4-1 MAXIMUM FUEL PLATE SURFACE TEMPERATURE AS A FUNCTION OF RECIPROCAL PERIOD. (16)

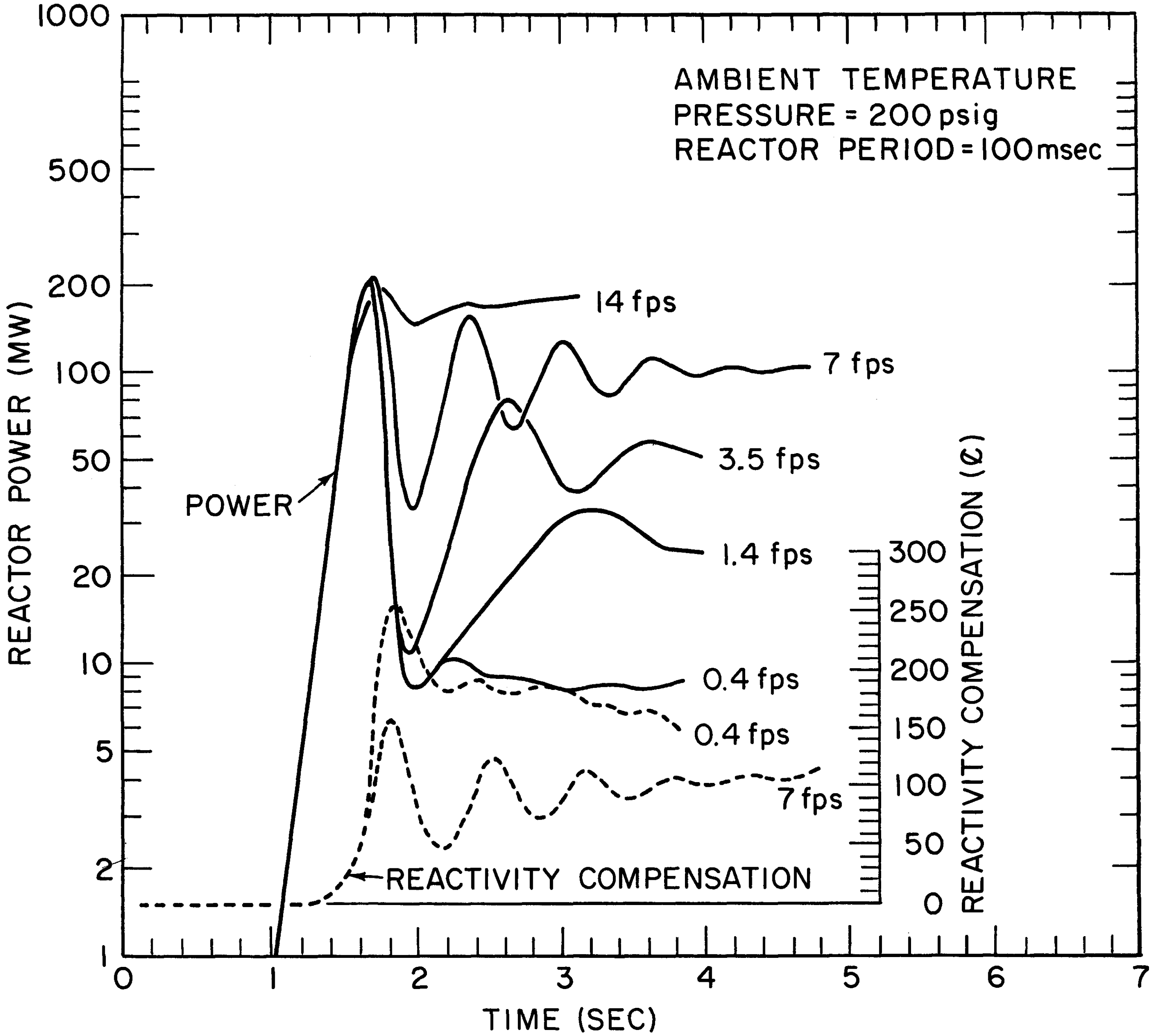

FIGURE III. 4-2 REACTOR POWER AND REACTIVITY COMPENSATION AS FUNCTIONS OF TIME FOR SPERT II 100-msec-PERIOD TRANSIENTS WITH DIFFERENT FLOW RATES. (17)

III.5 D_2O TANK LEVEL WITH REACTOR CRITICAL

Applicability

This specification applies to the level of D_2O in the reactor core tank which is permissible with the reactor critical.

Objective

To ensure that the temperature of the hottest fuel plate will not exceed 450°C (842°F) and to assure protection against loss of coolant.

Specification

The reactor shall not be made critical with the level of the heavy water in the main reactor tank lower than 18.0 in. below the overflow pipe level. The Limiting Safety System Setting shall be 5 inches below the overflow pipe.

Bases

If the reactor were to be made critical with little or no upper heavy water reflector, a sudden addition of more heavy water would lead to a large increase in reactivity. Hence, some limits must be placed upon a minimum upper reflector height.

Consideration of the safety limit for a maximum reactivity step addition (III.4) showed that a stepwise addition of 2.8% in reactivity could be tolerated without exceeding safe fuel plate temperatures. On this basis, a correlation can be made with past experiments conducted at the MITR which leads to the establishment of a safe heavy water level in the reactor vessel.

Experiments carried out by J. Lewins and C. Larson (Ref. (18)) during the startup of the MIT Reactor provide sufficient information to set this minimum allowable height. Reactivity worth of the top reflector was determined in a series of measurements on a clean core with varied shim height, a uniformly poisoned core, and a non-uniformly poisoned core. The latter two cores were designed to represent operating cores and other experiments have shown that they closely approximate the data. Table III.5-1 from p. 101 of the thesis report by C. Larson is reproduced here. This indicates quite clearly that a reactivity loss by lowering the level from overflow pipe level to 77.9 cm. (125.6 - 77.9 = 47.7 cm. loss in height) results in a loss of reactivity of 3880 mβ. In this reactor, β is worth 0.75% reactivity and, therefore, the loss of D_2O height to 80 cm. is clearly conservative for a Safety Limit. This is 45.5 cm. or 18.0 in. below the normal level. The estimated errors shown in the table, based on agreement of different sets of measurements and their repeatability, indicate that the accuracy is sufficient to assure the conservatism of the 18.0 in. figure.

The lower Limiting Safety System Setting of 5 in. below the overflow pipe is purely arbitrary and based on the fact that there is a tank level scram 4 in. below the overflow pipe level. It could as well be 8 in. since little reactivity change will occur by that time.

The level indicators on the tank ensure that if the levels drop below the set points safety actions including reactor scram will follow. Thus, a loss of coolant through a pipe rupture will be detected early and measures can be put into effect to minimize the consequences.

Table III.5-1 Reactivity Worth of Top Reflector by Critical Measurements

(a) Clean core, shim height varied

Heavy water level (cm.)	125.6	103.3	93.0	82.1
Reactivity (mβ)	0	-261 ± 5	-692 ± 14	-2760 ± 54

(b) Uniformly poisoned core

Heavy water level (cm.)	125.6	88.3	77.9	70.7
Reactivity (mβ)	0	-1290 ± 24	-3880 ± 80	-7240 ± 160

(c) Non-uniformly poisoned core

Heavy water level (cm.)	125.6	118.8	89.4	80.8	76.8
Reactivity (mβ)	0	-53 ± 3	-1081 ± 20	-2840 ± 60	-4150 ± 90

III.6. D_2 CONCENTRATION LIMIT

Applicability

This specification applies to the D_2 gas concentration in the helium gas cover blanket over the D_2O in the primary system.

Objective

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

Specification

The D_2 concentration in the helium blanket shall not exceed 6 volume percent. The Limiting Operating Value shall be 4%.

Bases

Recombination of the dissociated D_2 and O_2 is accomplished by continuously circulating the helium from above the reactor core 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 seventy degrees centigrade as measured at the middle of the reaction chamber.

In a thesis by John Nils Hanson, "Efficiency Study of the MITR Catalytic Recombiner," (22) it is shown that the recombination efficiency of the recombiner is 100% at 2 Mw and will be 100% at 5 Mw. In the experiments at 2 Mw, the recombiner was shut off for varying times up to 2 1/2 hours. The largest concentration of D_2 during these experiments was found to be 1.448%.

In a report, "Flammability of Deuterium in Oxygen-Helium Mixtures," issued by the Explosives Research Center of the Bureau of Mines (23), 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 percent O_2. The data in this report give the flammable concentration of D_2 at 25°C as 7.8 volume percent and 7.5 volume percent at 80°C. Extrapolation of these two points by a straight line approximation indicates a flammable concentration of 6.87 volume percent at a temperature of 200°C. These results are conservative since 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; so it can be concluded that combustion will not occur if the D_2 concentration is kept less than 6 volume percent.

The same report shows that even with deuterium-air mixtures of 30% D_2 plus 70% air the peak pressure reached in a mixture ignited in a 2-ft. sphere at 25°C at one atmosphere initial pressure was approximately 83 psig. This would seem to indicate that even in event of combustion of concentrations of deuterium far beyond that envisioned the pressures reached are not sufficiently high to rupture the primary containment in the region of the reactor vessel or major piping. This statement, while not amenable to direct proof, appears well-supported since the 2-ft.-diameter sphere represents a larger wall-to-wall distance than any vital section of the primary system. The most likely region for failure is the D_2O storage tank located in the basement. A rupture disk is located on this tank and even the total loss of this tank by an excessive pressure would not seriously harm the rest of the system. Therefore, it is concluded that this parameter, even in the extreme, may not pose a safety limit of any sort; although proof is not available.

The design pressure of the system is 40 psig, utilizing ASME Unfired Pressure Vessel conditions. Pressure surges of the order of twice design pressure are well within the capabilities which the system can withstand.

The accuracy of the experiments described in Ref. (23) is best indicated by a brief discussion of the data. All experimental mixtures which were flammable for the 80°C initial temperature tests were within 0.3-0.5 volume percent above the 7.5% deuterium mixture limit and those which were not flammable were within a similar band below the limit. Similar results are reported for the 25°C experiments. Thus, the set of experiments was carried out in such a way as to define quite accurately the limits of the region of flammability as a function of volume percent deuterium gas with from 60 to 90% helium present and the balance oxygen. The experiments were carried out at pressures of 0.5, 1.0, and 2.0 atmospheres and appear to be independent of pressure within that region. Since these experiments cover exactly the range of conditions and gaseous mixtures of interest here, they should be completely applicable.

The Limiting Operating Value for deuterium concentration in the helium cover gas was established at 4% primarily in order to provide a reasonable margin below the Safety Limit. In addition, it was necessary to set it high enough to ensure that too frequent audits were not necessary.

IV. LIMITING CONDITIONS FOR OPERATION

These conditions assure that Safety Limits are being properly observed, that vital equipment functions correctly, and that the reactor will operate in a safe condition.

IV.1. EMERGENCY COOLING REQUIREMENTS

Applicability

This specification applies to the emergency cooling system including the emergency cooling tank, valves, piping, transfer pumps, standby transfer pump, and fuel element spray heads.

Objective

To ensure that sufficient time is available for taking additional steps to cool the fuel elements subsequent to a loss of D_2O from the main reactor tank.

Specification

The emergency cooling tank shall be capable of providing for operation at power levels above 1 Mw a minimum of 20 minutes of emergency cooling flow over the fuel elements with a minimum total flow rate of 4.5 gal./min.

Bases

(A) As is shown below, a total flow rate of 4.5 gal./min. from the D_2O emergency cooling system is more than adequate to remove the decay heat of the hottest fuel element immediately following the loss of D_2O from the core tank. This assumes the reactor has scrammed, either from the low level tank scram, or from the loss of over 13 β associated with the loss of the moderator. Since the decay heat is a decreasing function with time, this flow rate is also more than adequate for cooling the core for times longer than 20 minutes. (The capacity of the emergency cooling tank is 175 gallons.)

1. Cooling of total core. Consider first the ability of the present emergency coolant to remove the decay heat from the core. Allow this coolant to completely evaporate. Assuming that the initial temperature in the coolant tank is 25°C, then the enthalpy change from liquid water to saturated steam is 404×10^3 kg-cal.

The decay heat, P, from fissions of U^{235} is given (24) as a fraction of the operating power, P_o, as:

$$\frac{P}{P_o} = 6.22 \times 10^{-2} \left(t^{-0.2} - (T+t)^{-0.2}\right)$$

where: t = decay time (sec.), and

T = irradiation time (sec.).

During a period τ after shutdown the total heat produced is:

$$Q = 6.22 \times 10^{-2} P_o \int_0^{\tau} \left(t^{-0.2} - (T+t)^{0.2}\right) dt$$

$$= 7.78 \times 10^{-2} P_o \left(\tau^{0.8} - (T+\tau)^{0.8} + T^{0.8}\right) \text{Mw-sec.}$$

If, to be conservative, the irradiation time is taken to be infinite, then for τ = 20 min. Q is 27.1×10^3 kg-cal. when P_o is 5 Mw.

Thus the capacity of the emergency coolant is more than adequate.

2. Cooling of center element. The decay heat of an individual element will depend on the operating power of that element. The less elements in the core the higher the individual element power must be. Also, the power generated in the element must be increased by heating from the gamma rays produced in surrounding elements.

Let F_γ be the energy production rate for gamma's and F_β be the production rate for beta's in a fuel rod.

(1) Based on calculations made at Harwell (25) for similar fuel elements, we assume that 100% of the beta production is absorbed in the element and 12% of the gamma's.

(2) Assume production rates are equal, as is stated on pp. 7-15 of Ref. (24).

$$F_\gamma = F_\beta = F \,.$$

The total energy production in an element is then:

$$2F = P \,, \text{ and } \quad F = \frac{P}{2} \,.$$

The total energy absorbed from the energy produced within the element is:

$$F_\beta + 0.12\, F_\gamma = 1.12\, F = \frac{1.12}{2}\, P = 0.56\, P$$

The rest of the energy, 0.44 P, escapes to surrounding elements.

(3) If the surrounding elements are assumed to produce equal power and are considered line sources, then the power reaching the central element from a neighboring element is 0.44 P (3/r) where r is in inches (1/r variation for line sources).

(4) Finally, assume 25% of the incident gamma radiation is absorbed. Then a surrounding element contributes 0.25 x 0.44 P (3/r) = 0.11 (3/r) to the energy absorbed in an element. Considering all the elements surrounding the central element, then P_a, the total absorbed energy,

$$P_a = 0.56\, P + 0.11\, P \sum_{i=1}^{N} \frac{3}{r_i} \,,$$

or

$$P_a = M_\gamma P = (0.56 + 0.11 \sum_{i=1}^{N} \frac{3}{r_i})P$$

where N_i is the number of elements in the ring.

For a 19-element core

$$M_\gamma = 0.56 + 0.11\ (2.82 + 2.72)$$
$$= 1.17$$

and for a 24-element core

$$M_\gamma = 0.56 + 0.11\ (2.82 + 2.72 + 0.864)$$
$$= 1.26\ .$$

In the 5 Mw Report (7), a 19-element core was considered with element power limited to 500 kw from reactivity considerations. Taking the burnup of this element to be 55 MWD (which is the maximum burnup to present), the heat absorption one second after shutdown including gamma ray absorption will be 484 kg-cal./min.

The flow rate through this element during the 20 min. of emergency cooling will be $\frac{175}{19 \times 20}$ gals./min. = 1.742 kg/min. Allowing this water to change to saturated steam would remove 1063 kg-cal./min., more than is being generated in the element.

(B) There are two methods of providing this cooling after the 20 min. of emergency cooling have expired, both utilizing the emergency cooling tank. In the first method the transfer pump (backed up by the standby transfer pump in case of electrical failure or other failures of the transfer pump) is used to supply D_2O to the emergency cooling tank. Suction can be taken from any part of the D_2O system or from the floor of the equipment room if the spillage has occurred there. In the second method light water is supplied to the emergency cooling tank from a city water supply through a quick-connect fitting. This second method is thus independent of any conceivable D_2O system failure causing loss of coolant from the core tank.

Since both methods involve relatively simple procedures and since either ensures a readily available adequate source of decay heat coolant, a 20-min. supply of emergency cooling water provides adequate protection to the core for this eventuality.

Since it has been shown on p. 692 of Ref. (26) that metal conduction to structural members combined with natural convective gas cooling is adequate for removing decay heat at power levels below 1 Mw, this specification applies only for power levels in excess of this amount.

The calculations in (24) and (25) provide adequate allowances for error and for spray variability within the element. No allowance has been taken for conduction of heat into the fuel element side plates and end boxes and thence into other structural members which p. 692 of Ref. (26) indicates is an important factor. No allowance is made for convective heat loss by gas or steam circulation. Omission of these factors is conservative.

IV.2. BUILDING CONTAINMENT AIR LEAK RATE

Applicability

This specification applies to the reactor building containment air leakage.

Objective

To ensure against the release of airborne radioactive effluent from the building in quantities endangering the general public in event of any accident within the building.

Specification

The air leak rate of the building containment shall be less than 2 percent of the containment air volume per day at 2 psig over-pressure when the ventilation dampers are closed. Full containment will be a requirement to permit reactor startup. An interlock shall be provided to assure that rods can only be withdrawn if no major containment leaks exist.

Bases

The possibility of the release of fission products from fuel elements has been discussed in the January 1956 MIT "Final Hazards Report," Appendix C; the October 7, 1960, (27) "Two Megawatt Report," Addendum to Appendix C (28); and the November 5, 1963, "Five Megawatt Report," changes to Appendix C. The conclusion reached is that the fission product release due to a fuel meltdown is an incredible accident.

As a justification for the leak rate specification of less than 2 percent of the containment volume per day at

2 psig over-pressure, a new evaluation has been made of the maximum exposure that could occur in the incredible event of fuel melt and fission product release.

Two mechanisms for the incredible accident which might melt fuel can be imagined:

(1) A sudden insertion of reactivity of such a large amount that the inherent shutdown mechanism of the void coefficient will not prevent fuel element meltdown.

(2) A sudden loss of all D_2O in the core during operation at 5 Mw with the emergency cooling system found inoperative in any form so that it will not prevent fuel melt due to fission product decay heating.

In the first case, the limitations on reactivity available in the core preclude the possibility of the reactivity insertion causing fuel meltdown. In the second case, only if the pipe from the emergency cooling tank were suddenly plugged or sheared at the time of the D_2O loss could the emergency cooling system be made inoperative. Even if the emergency cooling system were inoperative, it is not clear that the connecting aluminum adaptors will not conduct enough heat away from the elements to reduce the peak of the slowly rising temperatures even in the hottest element below the melting point of aluminum. Since both of these accidents are believed to be incredible and the second accident might not lead to fission product release, the first accident is assumed, therefore, to give a more conservative (largest) estimate of the largest concentration of fission products in the atmosphere of the containment shell.

The calculations presented in the 12/26/57 Addendum to Appendix B and C of the Hazards Report (MIT-5007) (27), as submitted to the AEC, shows that under very conservative assumptions the maximum possible exposure to radiation in one hour due to leakage of fission products out of the containment shell at the maximum permissible leak rate will be less than 1750 rem to the thyroid, 6.85 R total external β dose, and 0.6 mr gamma dose, after operation of the reactor at 1 Mw and complete fuel melting. Conservative, but more realistic, assumptions can be made which will reduce the magnitude of these calculated values.

First, experience with reactivity transients which lead to fuel melting would indicate that the core will not be completely melted. In the results of the SL-1 release, p. 680 of Ref. (26), it was found that 40% of the fission product inventory was involved in the melted fuel and, further, only 5 to 10% of the total fission product inventory escaped from the vessel. Thus it will be assumed that only 40% of the total fission product inventory is contained in the melted fuel plates, rather than 100% as used previously. It should be noted that the effects of after-heat in the MITR are much smaller than those in most power reactor cores being considered today and, therefore, the SL-1 experience is quite comparable to the hypothesized incredible accident being considered here. In addition, the fuel type is similar.

Second, the operating cylce of the MIT Reactor consists of approximately 100 hours of continuous operation and then a shutdown for the remainder of the week. Therefore, the I^{131} concentration does not reach the saturated equilibrium value, "A_{∞}". At the end of a long series of weekly 107-hr. irradiations the I^{131} activity is 0.7 A_{∞} or 70% of the saturated activity.

Third, in order for any of the building contents to escape, it is necessary for the pressure inside the building to rise above atmospheric pressure. Neither of the two accidents discussed above will lead directly to a significant building pressure rise. The reactor core is operated at a few inches of water pressure above atmospheric pressure, and a rupture of the reactor vessel will not release a large amount of energy to the atmosphere in the containment, in contrast to a large energy stored in a pressurized power reactor. Therefore, the possible increase in building pressure can only come from changes in the external atmospheric pressure after the containment is sealed. In the previous calculation it was assumed that a differential pressure of 1 psi above atmospheric could occur. However, such large changes in differential pressure have only been recorded in rare cases and occur over long time periods (more than a day). Since the exposure is calculated at the closest point which will be evacuated in a short time, it is reasonable to consider only short period changes in atmospheric pressure. According to the Boston Weather Bureau, a thunderstorm nose creates the shortest time pressure variations. The most marked example of such a pressure change since 1872 through 1946 occurred on June 8, 1946, when the pressure rose 0.19 in. of Hg, and then dropped 0.24 in. of Hg in a period of 1 1/2 hours. If the building had been sealed, the vacuum breakers would have opened during the pressure rise and the differential pressure would have risen to 0.24 in. of Hg during the atmospheric pressure drop. This occurrence and similar pressure drops recorded at the MITR indicate that such decreases are approximately linear with time. The average pressure differential for the worst case would then be about half maximum or 0.12 in. of Hg. It is then conservative to assume a maximum average differential pressure of 0.16 in. of Hg during the 1-hour exposure time.

The combination of these three items leads to a factor of:

$$\underbrace{\left(\frac{1}{0.4}\right)}_{\text{Fraction of core melt}} \times \underbrace{\left(\frac{1}{0.7}\right)}_{\text{Cycle operation}} \times \underbrace{\left(\frac{1}{0.0785}\right)}_{\text{Short time pressure rise}} = 45.7 \,,$$

by which the previous calculation of the thyroid dose should be reduced. (Note that the previous calculated dose was reported in units of rep with a conversion factor of 100 ergs/gm which is now designated as Rad, Rem = RBE x rad. However, for this case, RBE = 1.0 according to NBS Handbook 69. Hence, the previous value of 1750 is taken to be rem.)

After operation of the reactor at 5 Mw the calculated dose in the incredible event of a reactivity excursion leading to the release of fission products due to fuel elements melting and to the exposure at the boundary of the exclusion area to the maximum permissible leakage becomes:

$$D_I = \frac{5(1750)}{45.7} = 191 \text{ rem} \,,$$

to the thyroid due to I^{131}. Even if the reactor were to be operated on a continuous 24 hour a day schedule, the dose would not be greater than:

$$D_I = \frac{191}{0.7} = 273 \text{ rem} .$$

The external dose to a person standing in the leakage cloud due to beta or γ radiation from the cloud is calculated by assuming a continuous operation of the reactor for 180 days at 5 Mw prior to the fuel melting, and the above assumption concerning the fraction of the core melted and atmospheric pressure rise. By using the previous calculations, the external β dose becomes:

$$D_\beta = \frac{5(6.85)}{(45.7)(0.7)} = 1.07 \text{ rad in 1 hr.};$$

and the γ dose from the cloud at the maximum dose point becomes:

$$D_\gamma = \frac{5(0.565)}{(45.7)(0.7)} = 0.09 \text{ mrad in 1 hr.}$$

<u>Summary</u>. The dosage to critical organs from inhalation of all other radioactive fission products is much less than from iodine. It is concluded, therefore, that the gas-tight building surrounding the MIT Reactor provides adequate protection to persons outside the building against exposure due to fission products or inhalation of fission products even if fuel elements of the reactor should melt, an accident considered to be itself incredible.

An interlock must be satisfied requiring a minimum 0.10 in. water pressure differential (negative) in the building before a reactor startup can be conducted. This condition ensures that the building containment is not grossly violated at the time of startup. This same system shall provide a continuing alarm, if during operation the pressure differential is not satisfied.

IV.3. REACTOR CONTROL INSTRUMENTATION

Applicability

This specification applies to that instrumentation necessary for reactor control.

Objective

To insure that the operator has sufficient indication of power level, neutron flux level, D_2O flow, reactor tank level, and reactor outlet temperature.

Specification

A. The reactor shall not be brought critical or operated unless the following instrumentation is in operation:

(1) 2 neutron level channels
(2) 1 period channel
(3) Main reactor tank D_2O level indicator.

B. The reactor will not be operated above 200 kw unless there is indication of the following parameters in addition:

(1) Reactor D_2O outlet temperature
(2) Main D_2O flow.

C. Emergency power to operate the instruments indicated under IV.3-A1,A2,A3, and B1 above and also those specified under IV.4 shall be made available for at least one hour following a loss of normal power to the facility.

Bases

The neutron level channels and period channel provide indications of power level and change in power level during the approach to criticality and at low power levels. These instruments are therefore required at all power levels including subcritical operation. At power levels above 200 kw there is an additional need for flow and outlet temperature information to insure that the system is being operated within Operating Bound levels.

In practice, low power physics tests are usually done between 2 kw and 8 kw. To establish 200 kw as an upper limit for this type of operation, the calculated temperature rise as a function of time in the main tank with no main D_2O flow is shown below.

$$\frac{200 \text{ kw} \times 3414 \, \frac{\text{Btu}}{\text{kw-hr}} \times \frac{F^{\circ}\text{lb}}{\text{Btu}} \times \frac{5}{9} \frac{C^{\circ}}{F^{\circ}} \times \frac{1}{60} \frac{\text{hr.}}{\text{min.}}}{9.16 \, \frac{\text{lb}(D_2O)}{\text{gal}} \times 475 \text{ gal}} = 1.45 \; C^{\circ}/\text{min.}$$

Using this temperature rise and an intial temperature of $25^{\circ}C$, it would require approximately 50 min. for bulk boiling to be reached. Since this mode of operation will be used only for low power physics tests, the 50 min. provides ample time for observations and subsequent shutdown.

As the temperature of the coolant is raised from $25^{\circ}C$ to $100^{\circ}C$, the existing negative temperature coefficient will result in somewhat over 3.5 β in negative reactivity being added. Since no reactivity changes during low power physics tests will exceed a small fraction of β, the temperature of the bulk D_2O will rise only a few degrees before a low equilibrium power level is established. Thus, even in the event no shutdown occurs, the reactor is self-protecting.

At all times the MITR has present in the core valving which permits a convective flow within the primary tank in the event of loss of flow. This valving would permit convective flow in the less than 200 kw operation. In addition, work at Harwell (30) has shown that convective flow is induced within a core similar to the MITR core even if no such valving exists.

While the use of emergency power is probably not required in this reactor since loss of power automatically scrams the reactor and since coolant will still cover the core and melting will not ensue, none the less, the knowledge supplied to the operator that the reactor is properly secured and protected will insure an orderly procedure in all such cases. The choice of a minimum of one hour is based on providing reactor information during the period following scram and far enough beyond the emergency cooling cycle operation to insure that the core is receiving adequate subsequent cooling.

IV.4 RADIOACTIVE EFFLUENTS AND REQUIRED RADIATION MONITORS

Applicability

This specification applies to the levels of radioactive effluents released from the reactor site and specifies the radiation monitors needed to prevent release of radiation above these levels.

Objective

To ensure that operation of the MITR does not subject the public or MIT personnel to amounts of radiation above that allowed by 10CFR20.

Specification

1. The release of radioactive effluents from the reactor site will comply with all the provisions of Part 20, Title 10, Code of Federal Regulations with the following exemptions:
   a) A dilution factor of 3000 shall be applicable to the concentrations of gaseous effluents released from the stack.
   b) The gross quantity of tritium activity released to the sanitary sewer system shall not exceed 200 curies per year. All other radioactive material released to the sanitary sewer system shall not exceed one curie concentration.
2. During periods when ventilation is exhausted to the environment, a radiation monitor, which indicates in the control room, shall be operable and capable of detecting particulate and gaseous activity in the ventilation exhaust stream and of automatically closing the building vents.

Monthly checks will be made to ensure that the tritium concentration as the water vapor in the stack gaseous release is below the levels indicated in (1a) above.

3. A secondary system water monitor, which indicates in the control room, will be in operation whenever secondary cooling water is being circulated between the reactor building basement and the cooling towers. Sampling of this water for tritium content will be carried out at least once every 24 hours during those periods when the reactor is in normal operation with the main H_2O or D_2O pumps running.

4. At least one floor monitor capable of warning personnel on the reactor floor shall be in operation when the building is occupied.

Bases

Stack Gas Effluent Dilution Factor Basis:

The basic equations presented below were derived at MIT by D. Lanning and presented to the AEC April 24, 1959, as part of documentation backing up the MIT application for Amendment 3 of License R-37. (31) They were independently derived and presented by Hawkins and Nonhebel (32) and Moses, Strom and Carson (33).

The maximum ground level concentration from a point source at a height, h, above the ground as given by Sutton's formulation (34) is:

$$X_{max} = \frac{2Q}{\pi e} \frac{1}{\bar{u}h^2} \frac{\sigma z}{\sigma y} \qquad (1)$$

where Q = the discharge rate from the stack, (curies sec^{-1})

$\bar{u}$ = average wind speed (meters sec^{-1})

σ_z and σ_y = Gaussian standard deviations of the plume distribution.

The effective increase in stack height, ΔH, due to the discharge rate from the stack should be included. According to Holland, this is given by:

$$\Delta h = \frac{B}{\bar{u}} \tag{2}$$

where $B = 1.5 V_s d + (4 \times 10^{-5}) Q_h$

V_s = velocity of the stack effluent (meters sec^{-1})

d = stack diameter

Q_h = stack heat emission rate (cal sec^{-1})

Moses, Strom and Carson (33) provide evidence that the Holland formula is conservative.

Substitution of equation (2) in equation (1) gives:

$$X_{mas} = \frac{2Q}{\pi e u} \frac{\sigma z}{\sigma y} \frac{1}{(h + \frac{B}{\bar{u}})^2} \tag{3}$$

Differentiating X_{max} with respect to $\bar{u}$ and setting $\partial X_{max}/\partial \bar{u} = 0$ will give a maximum for X_{max} when $\bar{u} = \frac{B}{h}$.

Hawkins and Nonhebel (32) call this value of $\bar{u}$ the "critical wind speed."

Substituting this value of $\bar{u}$ in equation (3) gives

$$(X_{max})_{max} = \frac{Q}{2\pi e} \frac{\sigma z}{\sigma y} \frac{1}{Bh} \tag{4}$$

This concentration value is a double maximum representing a maximum with respect to horizontal distance from the source and the maximum with respect to wind speed. Further $Q = X_o F$ where X_o (curies/$meter^2$) is the concentration at the top of the stack and F (meters/sec^3) is the flow rate from the stack.

The stack effluent is normally not heated so Q_h in equation (2) is negligible. Equation (4) may now be written in terms of a concentration ratio

$$\frac{(X_{max})_{max}}{X_o} = (R_{max})_{max} = \frac{F}{2\pi e}\,\frac{1}{h(1.5V_s d)}\frac{\sigma z}{\sigma y} = \frac{F}{3\pi e h V_s d}\frac{\sigma z}{\sigma y}$$

But, by definition, $V_s = 4F/\pi d^2$. Therefore:

$$(R_{max})_{max} = \frac{d}{12eh}\,\frac{\sigma z}{\sigma y}$$

To compute $(R_{max})_{max}$ for the MITR one has

h = 50 ft., the vertical height between the top of the stack (150 ft. above ground) and the highest occupied level in neighboring buildings

d = 1.5 ft., the diameter of the stack at the top

$\frac{\sigma z}{\sigma y}$ = 1.0 conservatively

$$(R_{max})_{max} = \frac{1.5}{12(2.72)50} = \frac{1}{1080}$$

A study of the wind rose in the Cambridge area shows that the maximum time average concentration ratio will occur at a point east of the stack. The wind is in that quadrant no more than 30% of the time.

Therefore, it is clear that the time average concentration ratio will be substantially less than

$$0.3 \times \frac{1}{1087} = \frac{1}{3600}$$

The annual average concentration in the vicinity of the stack will be less than this value for the following reasons:

a) because the wind velocity changes in magnitude the downwind point of maximum concentration is not fixed as to distance from the stack as is implied by the equation.

b) in general the stack velocities and the R_{max} values will be such that the average wind speeds at the site will usually be higher than that value calculated to give the maximum concentration.

c) σ_z/σ_y is always less than unity in reality

d) the stack height above neighboring buildings in almost all directions and certainly to the east is greater than the 50 ft. value used.

In recent years there have been erected at MIT two buildings higher than the MITR stack. A third is planned.

It is of interest, therefore, to calculate a minimum dilution factor from the stack to these "tall" buildings.

The total dilution factor "D" for the effluent air between its origin at the stack top and the location considered is the product.

$$D = D_t \; D_e \; D_d \; D_g,$$

where D_t is due to decay in time of the radioactive effluent,

D_e is due to air entrainment in the plume as it leaves the stack at high velocity,

D_d is due to diffusion of the plume due to action of the wind,

D_g is due to variable direction of the wind.

The quantity D_t is calculated from the known half life of the radioactive isotope and the time it takes for the wind to move the effluent from the stack to the point in question.

The quantity D_e is obtained by the methods of Morton (35) and Scorer (36). They utilize conservation laws and the observed half angles of the plume to show that:

$$\frac{V}{V_o} = \frac{r_o}{r_e} \tag{1}$$

where V_o is the vertical speed of the effluent air at the stack orifice (22.9 m/sec),

V is the vertical speed of the effluent air at some distance beyond the stack outlet,

r_e is the radius of the plume of area $A = \pi r_e^2$ at the same distance from the stack outlet,

r_o is the radius of the stack orifice of area $A_o = \pi r_o^2$ (0.75 ft. = 0.229 meters).

Applying the conservation equation to the effluent gas gives

$$A_o V_o \rho_o = AV_\rho \tag{2}$$

where ρ_o and ρ are the density of the gas at the stack orifice and at the point where the velocity is V.

The entrainment dilution factor using equations (1) and (2) becomes

$$D_e = \frac{\rho_o}{\rho} = \frac{AV}{A_o V_o} = \frac{\pi r_e^2 r_o}{\pi r_o^2 r_e} = \frac{r_e}{r_o}$$

This relation is considered to hold until the vertical velocity equals the wind velocity. (and the plume radius is r_e) Then diffusion is assumed to start.

The quantity D_d can most easily be calculated by considering the ratio of the cross section of the plume at the beginning of the diffusion period and at the time it reaches the point in question. The work is simplified if the plume is considered to be diluted only in the horizontal plane. This is a conservative assumption.

If the plume horizontal width is equal to S at the building distance d, from the stack, then:

$$D_d = \frac{S}{2r_e}$$

The quantity D_g is calculated by considering it as the product of two terms, D_w, the dilution due to the fraction of the time the wind blows in the given octant, and D_r, the ratio of the arc length of the octant to the horizontal diffusion dimension S at the location considered a distance d from the stack.

$$D_g = D_w D_r = D_w \frac{1}{S} \frac{2\pi d}{8}$$

Combining these equations

$$D = D_\tau \; D_e \; D_d \; D_g = D_t \left(\frac{r_e}{r_o}\right)\left(\frac{S}{2r_e}\right)\left(\frac{2\pi d}{8S}\right) D_w = D_t \; D_w \frac{\pi d}{8r_o}$$

If the residual stack velocity remains greater than the wind velocity, D_d becomes unity, and S becomes $2r_e$. The result is numerically the same.

The three buildings and the results are:

	Direction from stack	Distance from stack	D_w	D
Earth Science Bldg.	E	590 (meters)	3.3	3,340 D_t
Eastgate	E	1035	3.3	5,860 D_t
Married Student Housing (Future)	SW	190	12	3,910 D_t

Thus, even if radioactive decay is neglected the average dilution factor to the nearest building will be greater than 3000. These calculations also indicate that addition of several additional high buildings should not cause any substantial change in these conclusions.

Tritium Factor Basis:

Leaks in the D_2O - H_2O heat exchanger systems resulting in discharge of tritiated water to the sanitary sewer system has always been considered a possibility at the MITR. It is believed that tritium discharged into the sewer at concentrations equal to or less than that provided by 10CFR20 presents no undue health hazard to the general public.

The average discharge of water from the secondary H_2O system of the reactor to the public environment per year is approximately 6×10^7 liters. Of this, approximately 10^7 liters is discharged as blowdown to the public sanitary sewer after dilution with an approximately equal volume of water coming from the Nuclear Engineering Building. The sewage from the site after further large dilutions by MIT and other users of this sewer goes into an MDC trunk sewer and is discharged at Deer Island to the ocean. The remainder of 5×10^7 liters is evaporated from the cooling tower.

Paragraph 20.106 of Part 20 of Title 10 CFR states in part:

(a) A licensee shall not possess, use, or transfer licensed material so as to release to an unrestricted area radioactive material in concentrations which exceed the limits specified in Appendix "B".

Table II of this part, except as authorized pursuant to 20.302 or paragraph (b) of this section. For purposes of this section, concentrations may be averaged over a period not greater than one year.

(b) An application for a license or amendment may include proposed limits higher than those specified in paragraph (a) of this section. The Commission will approve the proposed limits if the applicant demonstrates:

(1) That the applicant has made a reasonable effort to minimize the radioactivity contained in effluents to unrestricted areas; and

(2) That it is not likely that radioactive material discharged in the effluent would result in the exposure of an individual to concentrations of radioactive material in air or water exceeding the limits specified in Appendix "B", Table II of this part.

Any air release of tritium from the secondary water system falls under these paragraphs. The MPC for tritium in this case from Appendix B, Table II Column 1 is 2×10^{-7} μc/ml of air.

Further P. 20-303 states in part:

No licensee shall discharge licensed material into a sanitary sewerage system unless:

(a) It is readily soluble or dispersible in water; and

(b) The quantity of any licensed or other radioactive material released into the system by the licensee in any one day does not exceed the larger of subparagraphs (1) or (2) of this paragraph:

(1) The quantity which, if diluted by the average daily quantity of sewage released into the sewer by the licensee, will result in an average concentration equal to the limits specified in Appendix B, Table I Column 2 of this part; or

(2) Ten times the quantity of such material specified in Appendix C of this part (10 x 250 μc).

The release to the sanitary sewer of blowdown water from the secondary water system falls under these paragraphs. The MPC for tritium in this case from Appendix B, Table I Column 2 is 1×10^{-1} μc/ml of water.

Consider first the case of evaporation in the cooling tower. The chief meterologist of the Boston office of the U.S. Weather Bureau has provided MIT with approximate average monthly dew points for four reporting times daily for the past sixteen years. From these numbers, the annual average water vapor present in the Boston air is $\sim 5.7 \times 10^{-3}$ gms-H_2O/gm of air or $\sim 6.8 \times 10^{-6}$ ml H_2O/ml air. If one assumes that this water vapor is entirely due to evaporation of water containing tritium into average humidity air at the MPC of 20×10^{-8} μc/ml air, then the water must have had a concentration of approximately $20 \times 10^{-8}/6.8 \times 10^{-6} = 3 \times 10^{-8}$ curies/ml H_2O. Since approximately 5×10^{10} ml H_2O/year are discharged from the site via air release, this would permit the discharge of 5×10^{10} ml/year $\times 3 \times 10^{-8}$ curies/ml = 1500 curies/year without on the average exceeding acceptable concentration levels.

The situation is more complex than indicated here since the air at average humidity which enters the cooling tower will on the average have only undetectable amounts of tritium. It will leave the cooling tower with more water vapor per unit air volume than when it entered and may even be saturated. It will mix with other air and return to the general average humidity. In no case will the water vapor in the air be composed entirely of water evaporated from the cooling tower. The above calculation should therefore be quite conservative except, perhaps, in the region very close to the cooling tower.

Consider now the discharge to the sewers. On the average 10^{10} ml of blowdown water are discharged annually. If this is discharged at the MPC levels for sewers of 1×10^{-1} μc/ml including credit for the factor of 2 dilution which occurs at the reactor site (but no credit from other MIT installations feeding the same sewer) the total amount of tritium that could be discharged would be:

$$2 \times 10^{10} \text{ ml} \times 1 \times 10^{-7} \text{ curies/ml} = 2000 \text{ curies/year.}$$

By means of careful surveillance of the D_2O level in the primary system and by sampling daily during normal operation when the main D_2O and H_2O pumps are on, every effort will be made to detect heat exchanger leaks early. On the other hand the variability in tritium analysis counting accuracy and other factors will even under the best conditions permit some tritium to escape before it is detected.

With these considerations in mind, a value of 1/10 MPC or 200 curies/year for the sewer discharge has been selected as the maximum total tritium discharge in event of heat exchanger leaks. The escape of tritium, with the surveillance methods described above being used, should be readily observable before this quantity is exceeded. It also results in a maximum average concentration in the discharge sewer water of 2×10^{-8} curies/ml in the water which is well under the 3×10^{-8} curies/ml of H_2O required to ensure that the air concentration of tritium in the water vapor discharged from the cooling tower will be under the allowed values on the average.

Monitor Basis:

A negative pressure of 0.2 in. of H_2O is maintained in the MITR containment shell so that any air leakage at fittings will be into the containment shell. All ventilation enters and exits through ventilation headers. Both the inlet and outlet headers are fitted with filters, "Butterfly" dampers, emergency dampers, and fans. The monitors located in the ventilation exhaust main header are used to supply trip signals to seal the building and prevent the escape of radioactive effluents. The monitors located in the exhaust stack are to determine whether any high level release has actually taken place, and could also serve as backup monitors for building ventilation trips.

The plenum gaseous monitor is primarily used for detecting radioactive gaseous effluents but will detect high levels of particulate activity. The particulate monitor will detect radioactive gaseous effluents such as Argon 41 with a decreased efficiency. Thus, if the gaseous monitor is not operating or being repaired, the trip point on the particulate monitor can be lowered to provide adequate protection. Two monitors in the ventilation exhaust stack back up one another and, hence, adequate protection will still be present if one monitor should fail.

The secondary water monitor surveys the light water which passes through the cooling towers for the presence of radiation. It cannot detect tritium, but would detect the presence of N^{16} which would accompany any significant leakage of heavy water through the heat exchanger tubes into the light water during power operations. Therefore, it would immediately indicate a large D_2O-H_2O leak and the system would be shut down and isolated. If a D_2O-H_2O leak were to occur but be of such minor magnitude so as not to be detected by the water monitor, the periodic light water samples which are analyzed for tritium would indicate that a leak existed and the system could be isolated before excessive concentrations of tritium had developed in the system. During shutdown operations with the main D_2O and H_2O pumps off, for instance over weekends, the development of large leaks is unlikely and the main heat exchangers are isolated, thus ensuring a very small pressure difference even if a break should occur. The radioactive level in the H_2O would be very low in the event of a leak, but the background is also lower and Na^{24} in the leakage D_2O might still permit detection of the leak by the light water monitor which detects gamma rays.

At least one operating floor monitor is needed to insure that the general radiation levels in the occupied area around the reactor are within the limiting values established by 10CFR20.

IV.5. VARIABLE EXCESS REACTIVITY AND SHUTDOWN MARGIN

Applicability

This specification establishes the permissible variable excess reactivity and the required control rod worths.

Objective

To enable the reactor to be adequately shut down under all conditions.

Specification

1. Five shim rods shall be in operating condition and the main reactor tank at overflow before bringing the reactor to a critical condition.
2. The total variable excess reactivity above cold Xe free critical shall be less than the worth of the four least reactive shim rods.
3. An absorbing element shall be removed from the core for maintenance only if the core shall be left subcritical by at least the worth of the most reactive remaining shim rod.
4. The reactivity worth of the regulating rod connected to the automatic control system shall be less than 0.75% $\delta k/k$.
5. The reactivity worth of the limited dump of the top reflector shall be greater than the reactivity effect of the most reactive shim rod.

Definitions

1. The "cold Xenon free critical" MIT reactor is defined as a critical configuration in which:
   (a) The average D_2O temperature in the core is $10^{\circ}C$.
   (b) No fission product Xenon exists in the reactor.
   (c) The reactor is loaded for the beginning of an operating period.

(d) Variable reactivity effects such as sample changes which occur during normal operation shall be in the state in which they have the maximum positive reactivity.

2. Variable reactivity refers to the changes which may occur or vary during operation. It will include Xenon, fuel burnup (for one or more operating cycles), sample changes made in operation, changes in experiments during operation, etc.

Bases

Limiting the total excess reactivity to less than the worth of the four least reactive shim rods and with only five rods operating allows one rod to fail to drop while still completely shutting the reactor down.

Due to Xenon buildup after shutdown from equilibrium conditions, at least one more rod could be held up temporarily (up to 32 hrs at 5 Mw) and still hold the reactor subcritical.

In the six years of operation of the MITR, no rod has failed to drop on a scram signal. It is then considered unlikely that at least four rods out of five would not be inserted. In normal operation, six control rods would be in operation and only four rods out of six would have to drop.

The specification on removal of an absorbing element provides that the stuck rod criterion will always be met, even when one rod is removed for repair. Thus the reactor still would not go critical on the removal of a second element.

Since the movement of the regulating rod is governed by the automatic control system, a possibility exists of its accidental withdrawal. Its worth is therefore made a part of this Technical Specification. Although it is not deemed credible, the entire limiting worth of this regulating rod added to the reactivity worth allowed at the Limiting Operating Value (that amount which might be added 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) still would not exceed the Safety Limit of 2.8%. As indicated on the basis of Specification III.4 an addition of 2.8% would not endanger the integrity of the fuel elements. As a further precaution, at least an additional shim rod worth would be available from the limited dump of the D_2O reflector. This provides at least the equivalent of the stuck rod criterion safety margin in a completely independent back-up shutdown method.

As a general philosophical basis, the following stuck rod criterion is complied with at this reactor: "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." (26, p. 677)

IV.6 FUEL ELEMENT HANDLING AND STORAGE

Applicability

This specification applies to the operations of storing and handling fuel elements.

Objective

To ensure that fuel elements will be handled at all times in a manner to protect the health and safety of the reactor personnel involved and the public and to safeguard the elements.

Specification

1. Fresh fuel elements shall be stored in any of the following locations:
   - (a) In the reactor.
   - (b) Within the dry storage holes on the reactor top.
   - (c) In two approved storage vaults in the building adjacent to the containment building.
   - (d) Temporarily in rooms or sealed-off areas which can be locked for storage during experiments involving fuel.
2. Irradiated fuel element assemblies shall be stored in any of the following locations:
   - (a) In the reactor.
   - (b) In the dry storage holes on the reactor top.
   - (c) In the fuel storage tank in the basement of the reactor building.
   - (d) In the fuel element transfer flask or other proper shield within the controlled area.
3. Handling of fuel elements:
   Only one fuel element at a time shall be moved in or out of the reactor core. Not more than six of the MITR fuel elements shall be outside of the storage areas as designated in Items 1a,b,c and 2a,b,c, except

during the processes of receiving or shipping fuel from the site in approved containers. Records of fuel element transfers shall be maintained. Prior to transfering irradiated fuel from the reactor vessel to the transfer flask, the reactor shall have been shut down from power operation above 200 kw for a minimum period of twelve hours.

Bases

The principal problem in regard to fresh fuel elements is that of accidental criticality. The locations specified in 1a,b,c and 2a,b,c provide for complete criticality control. The reactor itself is of course shielded and appropriate written procedures assure that it is loaded properly. The dry storage holes in the reactor top are separate pipes poured in the concrete shielding and thus are isolated from one another neutronically. The fresh fuel storage vault and the spent fuel storage pit both have carefully designed geometric arrays to assure that criticality will not occur. The specification of no more than six elements outside of the designated storage areas of 1a,b,c and 2a,b,c assures that no criticality will occur elsewhere.

The chief additional problems with spent fuel are those of shielding personnel from the emitted fission product gamma rays and preventing melting from after-heat. The shielding requirement is met by utilizing a shielded transfer flask for movements and temporary storage and more permanent shielding as indicated in 2a,b,c,d. The requirement to prevent melting is met by specifying that 12 hours elapse between reactor shutdown and removal of the element from the core. Harwell (25) has shown by experiment that an element very similar to that used in the MITR and operated at a power of 500 kw/element will reach a maximum plate temperature of $\sim 450^{o}C$ if removed into non-circulating dry storage after 5 hours incore cooling and $360^{o}C$ after 12 hours incore cooling. Since the 500 kw element corresponds to the highest power element conceivable in the MITR at 5 Mw(t) this is deemed to provide sufficient conservatism.

IV.7 LIMITING CORE OPERATING CONDITIONS

Applicability

This specification applies to core conditions during operation of the MITR at power levels above 200 kw.

Objective

To ensure that the core is operated in a manner consistent with the assumptions used to evaluate the Safety Limits.

Specification

In addition to items stated elsewhere in the specifications:

1. The reactor shall not be brought critical unless all fuel elements and components such as thimbles, etc. are locked in position and all openings through the lower top shield plug sealed.
2. The reactor shall not be operated at power levels greater than 200 kw unless:
   a. All positions in the reactor plenum head are filled with either a sample assembly (or other unit with the same coolant flow characteristics) or a fuel element and there are no less than 19 fuel elements in the core.
   b. Each shim control rod is within 4.0 inches of a banked (average shim rod height) position.
   c. The rotary lid is latched in position.

Bases

The Safety Limits in Section III were derived by assuming that the D_2O flow is evenly distributed between fuel elements. There remains the possibility that the flow is not distributed evenly through the elements or that the power generation for a given element or elements varies more than allowed for in the calculations. There are two limiting conditions. One occurs when all the positions in the core contain fuel elements (30 elements) and the D_2O flow Operating Bound is set at 1800 gpm resulting in the minimum flow per element. The other limit occurs when the number of elements is reduced until the peak power in the central element exceeds that which was used for the derivation of the Safety Limit. The calculations were made for 19-element and 30-element cases; hence, the core should not be operated with less than 19 elements unless the calculations are again reviewed.

Further, the Safety Limit evaluations included the effect of banked shim rod height changes but unbalanced shim rod configurations might lead to higher power per element conditions then have been calculated; hence, the shim rods should be effectively banked within about 10%. Since the average differential reactivity worth of a single shim rod is approximately 1/6 β/inch, a deviation of $\pm$ 4.0 in. for a single rod will amount to about 2/3 β, or less than prompt critical. Since the total core operational excess reactivity is about 10 β, the effect of a single rod deviation of 4 in. will be less than 10% of the total banked shim worth.

The rotary lid acts as a biological shield on the reactor top. To facilitate the performance of various experiments placed in the core, the reactor may be operated at power levels below 200 kw with this shield removed. Based on measurements made under the lid the total dose rate at 200 kw is estimated to be approximately 2 Rem/hr. This dose rate is not in excess of those encountered during normal maintenance operations and adequate controls will be instituted during such experiments to prevent excessive personnel exposure.

V. OPERATIONAL SURVEILLANCE REQUIREMENTS

These conditions provide for surveillance to assure the reliable continuing performance and availability of vital instruments, equipment, and structures.

V.1. D_2O EMERGENCY COOLING FLOW

Applicability

This specification applies to the operation of the emergency cooling system.

Objective

To assure that the emergency cooling tank is full and ready to flow D_2O onto the fuel plates for a period of 20 minutes in the event of the accidental loss of D_2O from the main tank.

Specification

1. The emergency cooling tank will be checked before reactor startup to power levels of greater than 1 Mw if the reactor has been shut down for more than 24 hours, to prove that the "no overflow" alarm is operating properly, that D_2O is flowing into the emergency cooling tank, and that the valving is properly set and secured to ensure a continued flow at the proper rate.
2. Spray heads on each new fuel element shall be checked before insertion.
3. Checks will be made at least once every six months to insure that the valve setting which is secured will provide 20 minutes of cooling as the D_2O is drained from the emergency cooling tank.

Bases

It is believed that the specifications stipulated provide adequate assurance that the spray system will be operable in event it is required. The check of the system after any prolonged shutdown insures that D_2O is flowing into the emergency cooling tank (and, hence, out through the spray plates) and that the valving has not had its security breached. The six months check is a recalibration and appears sufficiently frequent on the basis of experience.

No total blockage of any spray head has been observed in over four years of operation of the spray system.

V.2. CONTAINMENT TESTING

Applicability

This specification applies to the operation of leak testing the reactor building containment.

Objective

To ensure the containment integrity of the building by periodic testing and inspection and to determine and record building leakage under test conditions.

Specification

1. An integral air leakage test on the reactor building containment will be conducted biennially with a maximum of 26 months time between tests. The entrance and exit quick operating valves will be inspected at least once every six months.

   A typical test will be conducted with the conditions of test pressure greater than 1 psig and less than 2 psig with blow-off leg set at 2 psig.

2. A test of the proper functioning of the independent vacuum relief breakers will be conducted annually with a maximum of 15 months between tests. New penetrations which may be installed will be checked for strength to resist the over-pressure and will be leak tested with soap bubble, freon, or helium methods.

Bases

The containment leakage specification sets limits on the amount of release of radioactive gas during any accident that could conceivably escape during an inconceivable but hypothesized accident. This specification is designed to give periodic proof that this containment will be always available.

The decision to make this a biennial test is based on practical experience utilizing annual tests in this facility since 1958.

On only one occasion out of eight has the facility failed to pass the leak rate tests at two pounds per square inch. In that one case, the margin by which the facility failed was very small (2.2% vs. 2%/24 hours) and was caused by wear and aging of the rubber gaskets on the intake and exhaust of the quick operating valves. Since each of these two valves is backed up by a second failsafe type closure, the facility could still have been maintained at 2 psig within its leak rate in event of an emergency. The provision to inspect the quick operating valves visually at six-month intervals recognizes that these valves are potential weak points.

The original design and specification called for a building to withstand 2 psig. The building contractor fabricated a building to withstand 2 psig, but stipulated that the guarantee was not valid at pressures over 2 psig. The design and the AEC license call for a building to withstand 2 psig. Therefore, the specification has been written to permit tests in the pressure range between 1 psig and 2 psig. Past experience has shown a negligible difference between the leakage predicted on a straight pressure difference calculational basis and that observed in the region of the pressure tests.

The vacuum breakers will be 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 is designed for pressure differentials of less than -0.10 psig (inside pressure minus outside pressure). While it is hard to envision a credible accident in which the containment was breached by an external over-pressure as a part of the accident, none the less, the vacuum breakers do provide additional assurance that external over-pressure cannot be a problem.

Past experience in the installation of additional penetrations has shown that any of the tests mentioned gives adequate assurance of leak tightness by utilizing the normally-available small negative difference in pressure between inside and outside. At 2 psig leaks of importance normally can be heard and are easily detected.

V.3. REACTOR INSTRUMENTATION SURVEILLANCE

Applicability

This specification applies to the operation of instrumentation whose surveillance is important to reactor safety.

Objective

To ensure the reliability and accuracy of the instrumentation important to safe operation of the reactor.

Specification

1. The following instrument functions will be tested or checked (and adjusted if required) each time before startup of the reactor if the reactor has been shut down more than 24 hours or if they have been repaired or deenergized:

   (a) Neutron level channels (minimum of two) - scram test
   (b) D_2O flow - scram test
   (c) Period channels (minimum of one) - scram test
   (d) D_2O tank level - scram test
   (e) D_2O outlet temperature - scram test
   (f) Helium circulation system and D_2O recombiner - operational check
   (g) Maximum security condition - major scram test (manual scram)
   (h) Emergency coolant system - operational check
   (i) Containment closure - ΔP interlock check
   (j) Stack, plenum, water, particulate, and area monitors - level set and trip point
   (k) Check voltage of emergency batteries.

2. The following instruments will be calibrated when initially installed, or any time a significant change in indication is noted, or at least annually:

   (a) Neutron level channels
   (b) Period channels

(c) D_2O flow instrumentation

(d) D_2O temperature detectors.

3. The calibration of the following radiation monitors will be checked initially and at least quarterly with a standard source:

   (a) Plenum monitor

   (b) Stack monitor

   (c) Particulate monitor

   (d) Water monitor

   (e) Area Monitor

Bases

An MITR operating cycle begins with an initial startup and closes with shutdown procedure. Each time before startup of the reactor, all scrams and important safety devices and interlocks are checked in accordance with a startup checklist. The shutdown checklist turns off all instruments not needed for shutdown cooling and safety parameters. Since some instrumentation is deenergized and other instrumentation may drift over a shutdown of several days it is deemed prudent to apply the specification above.

The 24-hour limit is established to permit restart after a shutdown for maintenance without the necessity to recalibrate all of the instruments. The limit of 24 hours was chosen on the basis of experience which indicates that the instrumentation used does not drift significantly or malfunction during that period if left energized and operating. In view of the overlapping indications available in every case and the knowledge which the operators have of the proper instrument behavior and readings, the 24-hour period is deemed not to affect significantly adversely the safety of the reactor.

If an instrument fails during operation and the reactor must be shut down to repair it and all other instruments are left in an energized and operating condition, only the affected instruments need be tested and checked before resuming operation within 24 hours for the reasons outlined above. If the repair takes longer than 24 hours, it is felt that the entire startup checklist must be done for the reasons outlined above.

All neutron level channels are given a relative calibration check each time the startup checklist is performed. The trip point is determined by the previous operating period chamber output current versus thermal power. This automatically compensates for component aging. Frequently, a cobalt foil irradiation is done as a check on process system thermal power instruments. The cobalt foil is always irradiated in the same position for the same amount of time, so that a relative neutron flux measurement is obtained. These operational checks ensure the reliability of the instruments, so that an annual overall calibration is sufficient to ensure safe, reliable operation. This has been verified by experience.

When a new radiation monitor is installed, an overall calibration is performed. A standard source is then placed by the detector and the instrument reading is recorded. The standard source check is repeated at least quarterly. An overall calibration is only considered necessary when the source check is not satisfactory.

Once the reactor is in operation, the proper function of these instruments is quite reliable and deviations from the normal are easily and quickly noted. The interaction of the parameters they measure makes it quite easy to double-check the performance of any one instrument. For instance, neutron level and period channels can be checked against one another. Or, to take a more complex case, a drop in D_2O tank level should give a level alarm and a no overflow alarm, then a scram, a separate indicator should show the drop, and a sight glass can be used to check it; and if the D_2O continues to drop without a scram reactivity will be affected and power will drop (unless control rods are pulled out to compensate), all neutron levels channels will show the drop, the period channels will show a negative period, the core outlet temperature should drop, and even the recombiner temperature will eventually fall as power drops and there is less deuterium to recombine.

V.4. RESPONSE TIME REQUIREMENTS AND SURVEILLANCE

Applicability

This specification applies to the response times for automatic safety devices and to the operation of measuring those times.

Objective

To ensure that the time response of the reactor safety devices is rapid enough to assure the safety of the reactor and its environs under all circumstances.

Specification

1. (a) The control rod trip time from initiation of the original input signal to the nuclear instrumentation, including the rod drop time, shall be $\leq$ 1.5 sec.
   (b) The time for the D_2O reflector to dump from the overflow pipe level to the dump level shall be $\leq$ 20 sec.
   (c) The time from initiation of the electronic signal for the radiation monitor trip in the plenum and including ventilation damper closing shall be less than the time for effluent air to flow from the radiation monitor to the damper.
2. The above requirements shall be checked one or more times each year. In particular, the period trip rod drop response time shall be checked utilizing simulated input periods of one second and 0.1 second.

Bases

The actual measured value for release time of the magnet is 50 msec and the measured value of the drop time is 430 msec. (Total release and drop time is thus $\leq$ 0.5 sec.)

The Bases for the Specification presented in III.4 showed that it might be necessary to assure that at least 2.8% or more in negative reactivity had been put into the core 1.5 seconds after initiation of the scram signal if the power oscillations observed in Spert II under flow conditions were not to be a problem. The 1.5-second requirement assured that the negative reactivity is added with an adequate safety margin. Figure E-2 of MITR-5007 shows that 2.8% in negative reactivity is added to the core by scram in approximately 0.21 seconds including the magnet release time, and 5% in 0.25 seconds.

The maximum D_2O reflector dump time shall be $\leq$ 20 seconds. The actual measured value is 11 seconds. If the reflector takes more than 20 seconds to dump, this would indicate the dump valve is not functioning properly and should be repaired.

The radiation monitor trip and damper closure time at various trip settings and count rates have been measured. Tests have been carried out in the past to measure the time required from initiation of a radiation monitor transient signal to the full closure of the ventilation damper. Tests have also been carried out to give assurance that the time for the passage of effluent air from the monitor to the damper is less than the closure time. The methods used for these tests are accurate enough to assure MIT that no activity from a puff burst will escape.

The choice of requiring that these checks be made annually is based on experience. No major deviations in any of the times specified in 1a, b, and c have been found since the reactor was first operated in 1958. In particular, it is difficult to conceive how the most important one, the control rod drop time could change materially since it is based on cutting off six electrical magnet currents and the independent free fall in open water and separately guided passages for each of the control rods. Therefore, for the rod drop test it is mostly the magnet release time that is being tested. This has been found to be quite constant and, therefore, could be tested at intervals of more than a year.

The D_2O reflector dump time (1b) and the damper closing time (1c) are somewhat more subject to variation as a function of use and, therefore, the frequency of test of one year has been based largely on (1b) and (1c).

V.5 RECOMBINER SURVEILLANCE

Applicability

This specification applies to the operation and surveillance of the D_2 catalytic recombiner system.

Objective

To ensure that recombination of the dissociated D_2 and O_2 is taking place.

Specification

1. Before increasing reactor power above 200 kw the following recombiner parameters shall be adjusted and checked:

   (a) The temperature in the middle of the recombiner $>50^{o}C$.
   (b) Recombiner flow rate >1.5 cfm and < 8 cfm.

2. If either of these parameters falls outside the above limits and cannot be restored within a two hour period, D_2 analysis shall be started or the reactor power will be reduced to < 200 kw.

3. As a part of normal operation at power the operator will assure himself at least once every two hours that the recombiner is functioning correctly.

Bases

Operating experience has shown that the recombiner operates most efficiently when operated in the range indicated in the above Specification. A rise in temperature in the recombiner is a result of the recombination process and is positive indication that the recombiner is performing its function. An efficiency study of the recombiner was done in a thesis by John Nils Hanson. For the results of Hanson's thesis, see the Bases of III.6 D_2 Concentration Limit.

VI. DESIGN FEATURES

VI.1 FUEL ELEMENT DESIGN LIMITATIONS

Applicability

This specification applies to design standards for fuel elements.

Objective

To ensure that the design of any fuel elements used in the reactor will provide adequate containment for the fission fragments, adequate and rapid heat transfer from the fuel through the clad to the water and adequate cooling of the element, and adequate boiling shutdown in event of a severe reactivity accident.

Specification

1. The fuel-bearing sections of the core shall consist of thin tubes or plates, each of which has a fuel section composed of uranium or UO_2 and aluminum clad by a layer of aluminum metal of not more than 0.025 in. thickness or less than 0.015 in.

2. The design of any element shall be such that in normal operation no bulk boiling shall occur in this or any other element in the core due to neutron flux peaking or D_2O flow distribution when the element is placed in any position in the core. This fact shall be ascertained by appropriate tests.

3. (a) Element designs shall be tested by prototype tests of dummy elements to see that they do not deform, disassemble or give indications of undue vibration at a flow rate of at least 12 ft. per second.

   (b) The element shall be mechanically locked in position in the reactor core during periods when the reactor is critical.

4. Any element design shall provide a minimum core averaged negative void coefficient of 170 mβ/liter of void in the active section of the element.
5. The design of any element shall be such that in a 19-element core operated normally (i.e., 19 elements or more and with shim rods banked) the calculated maximum fuel element plate temperature shall not exceed 450°C in event that any of the Safety Limits is reached.
6. All elements shall be so designed that, in a core containing 19 to 30 elements, they will operate in a stable flow regime within the limits set by the Technical Specifications on reactor power and outlet temperature.

Bases

Ref. (8) ("Process System Requirements of the MIT Reactor at Five Megawatts," W. R. Devoto) provides the basic equations and methods, and outlines the experiments upon which the design will be based.

The minimum clad thickness is based upon a series of MTR tests on elements with similar clad. These tests show that clad thicknesses of 5 or 10 mils permitted some fission products to be emitted. They also showed that 15-mil clad was completely adequate from the normal operating fission product retention viewpoint. Ref. (26), p. 672, shows that a thick clad increases the delay time for heat removal in event of a fast transient. Therefore, the clad should be as thin as possible while still remaining compatible with fission product retention requirements.

The selection of a no bulk boiling criterion assures that there will be no major fluctuations in reactor power due to the evolution and collapse of voids in the fuel and its feedback effects on reactivity. Also, as shown in III.1, III.2, and III.3, it has the effect of establishing a maximum heat flux per plate. This corresponds according to Ref. (7), as modified by the considerations of III.1, to the equation:

$$P_{PMAX} = 7.82 \times 10^{-3} \left(\frac{19}{N}\right)^{0.8} P_T \;. \qquad \text{(III.1-4a)}$$

Any fuel element should be designed for structural integrity and stability including possible vibrational effects. The design should assure that they will remain rigidly in place. Thus one specification requires that new types of elements be hydraulically tested at conditions more severe than normally encountered. The maximum flow velocity expected during 5 Mw operation is approximately 8 ft./sec.

As indicated in III.4, the principal shutdown mechanism which would terminate a severe nuclear transient is boiling. Thus the quantity of reactivity worth which can be obtained to shut down the reactor in such an event is an important part of the core transient behavior. Thus a specification of the void coefficient and total available void volume in each element is important. The choice of this particular specification insures that future fuel elements will comply with the Safety Limit set forth in III.4.

The ultimate safety goal of fuel element design remains to ensure that the 450°C plate surface temperature is not exceeded. This value was chosen as indicated in the Introduction because aluminum melts at about 660°C and its strength deteriorates significantly with temperatures above

450°C. Harwell reports that they have observed severe blistering of irradiated fuel under prolonged exposures to temperatures of 550°C with the onset of the phenomenon at about 500°C (25).

The design of any fuel element must also ensure that no flow instabilities or burnout can occur in thermal-hydraulic regimes which could conceivably be reached in normal or abnormal operation. Therefore, a specification has been set up so as to make the design and characteristics of any new fuel elements be self-consistent with those now in use.

The specifications set forth above make the thermal, hydraulic, and nuclear characteristics of any allowable fuel element design consistent with the Safety Limits described in these Specifications. Therefore, the utilization of any fuel element complying with these Specifications does not constitute an unreviewed safety problem.

VII. ADMINISTRATIVE REQUIREMENTS

VII. ADMINISTRATIVE REQUIREMENTS

1. Organization

(a) The MITR shall be administratively controlled by the Administration of the Massachusetts Institute of Technology. The chain of administrative authority and responsibility includes successively the President of the Institute, the Dean of the School of Engineering, the Chairman of the Department of Nuclear Engineering, and the Reactor Director.

(b) The Reactor Director, a member of the Department of Nuclear Engineering, shall be responsible for overall direction and operation of the reactor facility and control of the reactor fuel.

(c) The Reactor Operations Superintendent shall be responsible to the Director for the operation, maintenance, and refueling of the reactor and for it associated facilities.

(d) The MIT Assistant Medical Director in charge of the Occupational Medical Service shall be responsible for radiation protection at the MITR and its facilities. The MIT Assistant Medical Director reports through the Medical Director to the President's Office. He shall provide for review and approval of all types of experiments and shall provide for monitoring of operations and conduct of experiments with respect to radiological safety.

(e) There shall be a local Reactor Safeguards Committee which shall be responsible to the Administration of

MIT. The Committee shall review and approve from the safety viewpoint: (1) the principles and administration of operating and radiation protection procedures, (2) proposed modifications to the reactor affecting its safety, and (3) general types of experiments and special specific experiments for each of the experimental facilities of the reactor and proposed modifications to such experiments. The Committee shall conduct or have conducted periodic audits of operations, equipment performance, logs, and procedures.

(f) The reactor program for medical applications shall be reviewed by the MIT Reactor Biomedical Advisory Committee. This Committee shall be responsible to the Administration of MIT and shall review and approve the biomedical experiments.

(g) General types of experiments and special specific experiments for each of the experimental facilities of the reactor as approved by the Safeguards Committee or by the Reactor Biomedical Advisory Committee shall also be reviewed and approved prior to installation in the reactor by the Reactor Director and the MIT Assistant Medical Director if not members of the approving Committee.

(h) Each experiment shall be reviewed and approved by a supervisor of the reactor operations staff and by a responsible member of the radiological safety staff prior to insertion into the reactor.

(i) During the intervals between periodic audits noted in 1(e) the Reactor Director shall provide for continuing review of operations, procedures, and equipment performance.

(j) The results of reviews, approvals or recommendations, and the bases for such approvals or recommendations by each of the groups mentioned in paragraphs (d), (e), (f), (g), (h), and (i) shall be documented and kept as a part of the MIT Reactor records.

(k) Regularly assigned members of the organization described above, if unavailable for normal functions, may be temporarily replaced by qualified substitutes by written authorization of the regularly assigned member or his superior.

2. Procedures for Operation

(a) Detailed written procedures and check sheets (i.e., manipulation instructions) for operation of the reactor and its supporting facilities, maintenance operations, and emergency operations and radiation protection shall be provided and shall be in conformance with these Technical Specifications.

(b) All initial detailed written procedures and check sheets shall be prepared by the reactor operations staff and shall be reviewed and approved in writing by the Reactor Operations Superintendent and the Reactor Director before being put into effect. This review shall be made in such a way as to ensure conformity with these Technical Specifications and the principles and administration of operating and radiation protection procedures as set down by the MIT Reactor Safeguards Committee. Subsequent changes or modifications to the initial detailed written procedures which do not alter the intent of the initial procedures shall be approved by the Reactor Operations Superintendent.

(c) Written procedures shall include, but not be limited to, the following:

Hydraulic and nuclear startup or shutdown of the reactor,

Fuel element charging to and discharging from the reactor,

Replacement of the control rods,

Installation and operation of experiments,

Performance of normal major maintenance on system components, and

Action to be taken in event of abnormal conditions of the reactor plant.

3. Reactor Operation

Members of the operating staff shall each be familiar with each procedure for which he has responsibility. A complete and current set of procedures, for both normal and abnormal conditions, shall be maintained in the reactor control room. A personal set of radiation protection procedures shall be issued to each person working in the MITR facility. Upon occurrence of abnormal conditions in controls, safety systems, auxiliary systems, or experiments, or if a Limiting Safety System Setting or Limiting Operating Value is exceeded, action shall be taken immediately to ensure the safety of the facility and determine the cause of the abnormal behavior.

The staff shall conduct drills at sufficient frequency to insure proficiency in emergency procedures.

4. Logs and Records

Logs of operation and maintenance shall be maintained.

(a) The operations log shall have recorded:

(1) routinely pertinent data regarding system operation,

(2) actions of operators and experimenters, and

(3) details of any abnormalities occurring and actions taken thereon

(b) The maintenance log shall have recorded:

(1) routine maintenance component replacement and calibration,

(2) equipment failures, and

(3) replacement of major items of equipment.

5. Changes

Changes to facility equipment (other than replacement of components or parts) shall be reviewed and approved by the Reactor Director and at least one other qualified person as he may designate. Results of such reviews, with bases therefore shall be documented and maintained as part of the MIT Reactor records.

6. Operating Limitations

(a) Whenever fuel is being positioned in the reactor grid plate or any operation is being performed that could result in the release of radioactivity or create a change in radiation levels or affect reactivity, the reactor and its supporting facilities shall be maintained on an operational basis as described in these Technical Specifications.

(b) Whenever the reactor is critical, or its control rods can be raised or are being manipulated, there shall be a minimum crew of two men, one of whom shall be in charge and shall bear responsibility for the safe operation of the facility. He shall have a valid Senior Operator's license.

(c) Whenever the reactor is critical or undergoing an operation which may change its reactivity, the control room shall be attended by a licensed operator and the reactor instrumentation shall be in operation and monitored by the operator. Manipulation of any components, mechanisms, or apparatus conducted outside the control room which may affect safety of any reactor system or experiment shall be conducted with the knowledge of the reactor operator.

7. <u>Actions to be Taken in Event Technical Specification Limits Have Been Exceeded or Stated Conditions Are Not Met</u>

(a) If any of the Safety Limits of Technical Specifications III.1 through III.6 are exceeded, the reactor immediately shall be placed in a condition of maximum security, and the situation shall be reported to the Division of Compliance, Region I. Reactor operation shall not be resumed until regulatory approval has been received.

(b) In the event any condition as stated in Technical Specifications IV.1 through VI.1 is not met,

(1) if the reactor is in operation when the condition is discovered, either execution of approved emergency procedures for temporary continuance of operation or reduction in power or shutdown, as appropriate, shall be accomplished until the condition is rectified, or

(2) if the condition is discovered while the reactor is shut down, operation shall not be resumed until the condition has been rectified.

(c) If any Limiting Safety System setting, Limiting Operating Value, or Limiting Condition of Operation is exceeded, the condition shall be reviewed by the Reactor Director and such other persons as he may designate. The review and basis for actions taken as a result thereof shall be documented and maintained as part of the MIT Reactor records.

(d) The responsibility for adherence to limits and conditions set forth in this license shall rest with the MIT Administration. Any instance of failure to observe administrative requirements set forth herein shall be reviewed by the MIT Administration and the result of such review and actions taken, with bases therefore shall be documented and maintained as part of the MIT records.

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