NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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Activation Analysis
Coolant Chemistry
Nuclear Medicine
Reactor Engineering

22 August 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: License Amendment for MIT Research Reactor, License No. R-37, Docket No. 50-20, "Technical Specification No. 6.7 – Experiments Involving In-Core Irradiation of Fissile Materials"

Dear Sir or Madam:

On November 21, 2001, the Massachusetts Institute of Technology submitted the above referenced license amendment. That request has now been reviewed following discussion comments, and we believe that the amendment can be improved by further specifying more of the planned safety aspects. Accordingly, the amendment request submitted as of November 21, 2001, is hereby withdrawn and the enclosed request is submitted.

Please contact either of the undersigned should further information be required.

Sincerely,

Lin-Wen Hu, Ph.D.
Utilization Engineer

John A. Bernard, Ph.D
Director

I declare under penalty of perjury that the foregoing is true and correct.
Executed 8-23-02 ____________
Date Signature

cc: USNRC - Senior Project Manager
NRR/ONDD

USNRC - Region 1 – Project Scientist,
Effluents Radiation Protection Section (ERPS)

A020

Safety Review #-0-01-11

A. TS #6.1.7 "Radioactive Releases"

The current wording of TS #6.1.7 (b) is overly conservative. It states that "the total radioactive materials inventory of an experiment or credibly coupled experiments shall be limited such that the dose in unrestricted areas resulting from release of this inventory at its calculated maximum value shall not exceed that of the Design Basis Accident." In reality, release of radioactive materials is limited by both the initiating event and its associated release path. The current wording implies the assumption that the maximum radioactive inventory is released regardless of the initiating event and its release path.

The proposed wording for the revised TS #6.1.7 is attached. The new wording sets the limit of offsite dose resulting from an experiment malfunction to be that of the 10 CFR 20.

B. TS #6.7 "Experiments Involving In-Core Irradiation of Fissile Materials"

1. Description of Change

The MITR currently does not have a technical specification (TS) for in-core irradiation of fissile materials. The purpose of the proposed TS amendment is to provide general requirements for in-core irradiation of fissile materials. Each new type of fissile material irradiation experiment will be analyzed and reviewed/approved individually by the MIT Reactor Safeguards Committee (MITRSC) because the reactivity effects, heat transfer characteristics, etc., can vary significantly depending on the type of fissile material, its enrichment, and its geometry.

2. Limitations on In-Core Irradiation of Fissile Materials for Research Reactors

The MITR is licensed as a research reactor. The limitations on in-core irradiation of fissile materials for research reactors are imposed by the definition of a research reactor. Code of Federal Regulations 10 Part 50.2 defines a non-power reactor as a research or test reactor licensed under 10 CFR 50.21(c) or 50.22 for research and development. A test facility is defined in 10 CFR 50.2 as a nuclear reactor for which "...an application has been filed for a license authorizing operation at: (1) a thermal power level in excess of 10 megawatts; or (2) a thermal power level in excess of 1 megawatt, if the reactor is to contain: (i) a circulating loop through the core in which the applicant proposes to conduct fuel experiments; or (ii) a liquid fuel loading; or (iii) an experimental facility in the core in excess of 16 square inches in cross-section." Therefore, it is possible to irradiate fissile materials in the MITR core if the conditions of provisions 10 CFR 50.2(2) (i) through (iii) are avoided.

3. Other Limits Specific to MITR

Other limits on the in-core irradiation of fissile materials specific to the MITR in-core experiments are experiment reactivity worth limit and onset of nucleate boiling (ONB). Additional requirements such as weekly sampling of cover gas in the void space and over-temperature automatic reactor scram provide redundant protection against a potential malfunction of the fissile materials irradiation experiments. The limit on U-235 content in a fissile materials irradiation experiment is derived from the Design Basis Accident (DBA) of the reactor. The effect of actinides, which are produced from U-238, on off-site dose is analyzed. It is concluded that a limit on the initial amount of U-238 is not required.

3.1 Off-Site Dose during DBA

The Design Basis Accident (DBA) chosen for the MITR-II assumes a blockage of five coolant flow channels that results in four fuel plates completely melted. (Note: It is highly unlikely that a large piece of foreign material would block the coolant channels completely because it would have to have fallen through the upper grid hold down mechanism when a fuel element was removed.) Release of the fission products to the atmosphere was calculated assuming the fission product buildup achieved saturation Off-site dose to the general public was then calculated from the released fission products. The maximum amount of fissile materials that can be accommodated in a fissile material experiment should result in a maximum fission product release below that of the DBA. Hence the scope of the SAR is not altered by the in-core fissile material irradiation. Using a simple approximation based on the U-235 content, the maximum amount of U-235 would be 506 grams (mass of U-235 per element) x 4 (plates) / 15 (plates per element) = 135 grams. A limit of the total initial amount of 100 grams U-235 is conservatively chosen.

The off-site whole body dose resulting from actinides is calculated. Details of the calculation are included in the attached memo. Actinides are produced when U-238 is irradiated. The off-site whole body dose from actinides is calculated to be 2 mrem per kilogram of initial U-238. The maximum initial amount of U-238, which is set by the total off-site dose from both fission product and actinide releases of the fission materials experiment, is calculated to be 31 kilograms. This amount is significantly higher than that of natural uranium that contains 100 grams of U-235, 0.1 kgU-235×(0.993/0.007)= 14.2 kgU-238. Therefore, a limit on the initial amount of U-238 is not required.

Simultaneous occurrence of the DBA and failure of an in-core fissile material experiment are not considered credible because coolant channel blockage of both a fissile material experiment and a fuel element is extremely unlikely.

3.2 Reactivity Effect

TS #6 1 "General Experiment Criteria" limits the maximum reactivity worth of a secured experiment to 1.8% $\Delta K/K$ (or 2.3 beta). It was previously calculated that addition of one gram of U-235 (93% enriched) in the A-ring would result in a positive reactivity insertion of 8.29 mbeta, which was obtained in refueling calculations Therefore, the maximum amount of U-235 that can be added in the A-ring would be 2300 mbeta / 8.29 mbeta/gram = 277 grams of U-235. Since the reactivity effect could vary significantly by the type of material, geometry etc., each type of fissile material irradiation should be analyzed and reviewed.

3.3 Onset of Nucleate Boiling

The limit on the thermal power generated from a fissile material experiment is primarily imposed by the onset of nucleate boiling (ONB) which is one of the criteria in TS #6.1 "General Experiment Criteria". Each fissile material irradiation experiment facility design will be analyzed and reviewed to ensure that ONB would not occur during steady-state operation However, 200 kW, which is the average thermal power per element, is used to set an upper bound for any fissile material irradiation experiment.

3.4 Over-Temperature Automatic Scram

The inner barrier of the double encapsulation is monitored for over-temperature. The scram setpoint is experiment-dependent and will be chosen to avoid rapid mechanical and/or chemical degradation of the barrier. The scram setpoint will be documented in the safety review that is required by for each in-core fissile materials irradiation experiment.

4. Off-Site Dose Calculations for Fission Product Gases Release

Fission product gas release is an unlikely accident scenario because:

a) As shown in Section 3, the proposed technical specification requires weekly sampling of cover gas of a fissile materials irradiation experiment when the reactor power is at 100 kW or higher. This will detect any potential leakage of fission product gases from the barriers of a fissile materials irradiation experiment.

b) The core purge monitor will detect a radioactivity release that is higher than normal. The core purge monitor reading is taken hourly by the on-console operator. A higher than normal reading will be reported to the Reactor Radiation Protection staff and sampling of the core purge is required to identify the cause. A "High Core Purge Monitor" alarm will alert the operator if the reading exceeds the setpoint of 100 kcpm.

c) If the fission gas release is caused by high temperature of the fissile materials/barriers, an over-temperature alarm will automatically scram the reactor and thus minimize all fission products release.

An analysis is performed assuming that, as a result of multiple failures, all fission product gases produced from a fissile materials irradiation experiment are released through the reactor ventilation system. The fission product gases analyzed here are the noble gas nuclides including Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, and Xe-135. The fission product gas inventory is assumed to be at equilibrium at 100 kW (maximum allowable power of a fissile materials irradiation experiment) and is released within one week. The analysis concludes that a) the core purge monitor should detect a higher reading, 44 kcpm over background, if the fission product gases were to escape the barriers, an increase equivalent to approximately twice that of a normal background reading, and b) if the entire fission product gas inventory leaks from a fissile materials irradiation experiment and is released to the atmosphere through the stack, the total inhalation dose is calculated to be about 17 mrem. The inhalation dose of 17 mrem is much lower than the 100 mrem annual limit for general public defined by 10 CFR 20.

There will be no additional thyroid dose because none of the fission product gases affect the thyroid.

5. <u>Proposed Wording for TS #6.7</u>

See attached.

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Activation Analysis
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MEMORANDUM

TO: MITR Safety Review #O-01-11

FROM: Lin-Wen Hu

DATE: October 25, 2001

RE: Actinides Off-Site Dose Calculations During DBA

1. One question that was brought up during the review of TS 6.7 " Experiments Involving Irradiation of Fissile Materials" was the amount of U-238 that would be allowed in addition to the fissile material (i.e., U-235). Irradiation of U-238 produces a very small percentage of fissions (fast fission), however, the activities of the actinides, especially the alpha-emitters, may contribute to additional whole body dose during the design basis accident.

2. The actinides that are produced during irradiation are summarized in the table below. The activities at end of cycle (EOC) correspond to an initial U-238 loading of 330 grams and a burnup of 100 MWd/kg iHM (iHM: initial heavy metal), or 520 effective full power days of irradiation in the MITR at 5 MW. Most of the actinides (the ones with shorter half-lives) reach equilibrium quickly and therefore higher burnup will not affect these activities. For actinides with longer half-lives, i.e. Pu-238, Pu-239, Pu-240, and Pu-241, their activities increase with burnup. Activities of the Pu nuclides are calculated from extrapolating data at lower burnups, which were calculated using MONTEBURN by Pavel Hejzlar.

Actinides	Half-Life	Decay Mode	Activity EOC (Ci)	Inhalation ALI (μCi)
U237	6.75 days	β-decay to Np237	2.41×10^2	2×10^3
U239	23.5 minutes	β-decay to Np239	4.60×10^3	2×10^5
Np238	2.12 days	β-decay to Pu238	35.6	60
Np239	2.36 days	β-decay to Pu239	4.58×10^3	2×10^3
Pu238	87.7 yrs	α-decay to U234	2.735	7×10^{-3}
Pu239	24110 yrs	α-decay to U235	0.313	6×10^{-3}
Pu240	6564 yrs	α-decay to U236	0.315	6×10^{-3}
Pu241	14.35 yrs	β-decay to Am241	129.3	0.3
Cm242	162.8 days	α-decay to Pu238	0.777	0.3

3. The off-site whole body dose contributed by the actinides is calculated by using the thyroid dose components as a reference. The thyroid dose was previously calculated by Q. Li (1998) using atmospheric release models which take into account both release from the pressure relief system (stack) and release from containment leakage. The containment leakage was concluded to be the dominant term for the atmospheric release at the reactor boundary. The thyroid dose during 2 hours of release was calculated to be 135 mrem for a reactor power of 6 MW. These results are summarized in the MITR-III SAR. The ratio, R, of the off-site whole body dose resulting from the actinides to the previously determined off-site thyroid dose can be calculated using the following equation:

$$R = \frac{\sum_i \frac{A_i \times D_{wb}}{ALI_i} \times \left(\bar{F}_f \cdot F_d \cdot F_{RCS}\right)_i}{\sum_j \frac{A_j \times D_{thyroid}}{ALI_j} \times \left(\bar{F}_f \cdot F_d \cdot F_{RCS}\right)_j} \tag{1}$$

where

- i denotes actinide isotopes U-237, U-239, Np-238, Np-239, Pu-238, Pu-239, Pu-240, Pu-241, and Cm-242,
- j denotes thyroid-seeking nuclides I-131, I-132, I-133, I-134, I-135, and Te-132,
- A is activity,
- ALI is the annual limit on intake; the inhalation limits for whole body are used for the actinides, and the inhalation limits for thyroid are used for I and Te;
- D_{wb} is the annual dose limit for whole body (5 rem),
- $D_{thyroid}$ is the annual dose limit for thyroid (50 rem),
- $\bar{F}_f$ is 1 for containment leakage (no charcoal filter),
- F_d is natural depletion in containment, and
- F_{RCS} is release fraction from fuel to reactor coolant system.

The whole body dose resulting from actinides can then be calculated using the following equation:

$$D_{wb,actinides} = R \times D_{thyroid,DBA} \tag{2}$$

where

$D_{thyroid,DBA}$ is the calculated thyroid dose for DBA

The release fractions for Iodine, Tellurium and actinides are summarized as following:

	$\bar{F}_f$	F_d	F_{RCS}
Iodine	1	0.3	0.9
Tellurium	1	1	0.23
Actinides	1	1	1e-4*

* The RCS release fraction for actinides is obtained from severe fuel damage tests [NUREG/CP-0090, June 1988].

4. The off-site whole body dose resulting from irradiation of U-238 is calculated using Eqs (1) and (2). Using a burnup of 100 MWd/kgiHM, the off-site whole body dose is calculated to be 0.002 mrem per gram of initial U-238, or 2 mrem per kg of initial U-238. Assuming 100 grams U-235 is used in the fissile material irradiation (proposed TS limit), the off-site whole body and thyroid doses resulting from complete damage of the material can be calculated from those of the reactor's DBA. The second column in the following table contains the off-site dose calculated by Q. Li for the reactor's DBA. The off-site doses in the third column are calculated for 100 grams of U-235 (without U-238), which is the proposed upper limit of the fissile material experiment.

Component of the Dose	Dose (mrem) (a)	
	DBA (4 plates melted) 21 m (a)	DBA (fissile mat. exp.) 21 m (a)
Whole body:		
Containment Leakage	12	9
Steel Dome Penetration	25	19
Shadow Shield Penetration	21	16
Air Scattering	75	56
Steel Scattering	114	84
Total	**247**	**184**
Thyroid:		
Containment Leakage	**134**	**99**

(a) Nearest point of public occupancy

(b) Calculation assumes that radiation emergency plan for protection of the public will be implemented in less than two hours.

Thus, the maximum allowed U-238 can be determined from $\frac{247\text{mrem} - 184\text{mrem}}{2\text{mrem} / \text{kgU238}} = 31.5$ kgU238. This amount is significantly higher than that of natural uranium with 100 grams of U235, 0.1 kgU-235×(0.993/0.007)= 14.2 kgU-238. Therefore, a limit on initial amount of U-238 is not required.

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MEMORANDUM

TO: MITR Safety Review #O-01-11

FROM: Lin-Wen Hu

DATE: August 14, 2002

RE: Off-Site Dose Calculations for Fission Product Gases Release

1. Fission product gas release is an unlikely accident scenario because

    a) TS#6.7 requires weekly sampling of cover gas of a fissile materials irradiation experiment when the reactor power is at 100 kW or higher. This will detect any potential leakage of fission product gases from the barrier of a fissile materials experiment.

    b) The core purge monitor will detect a radioactivity release that is significantly higher than normal. The core purge monitor reading is taken hourly by the on-console operator. A higher than normal reading will be reported to the Reactor Radiation Protection staff and sampling of the core purge is required to identify the cause. A "High Core Purge Monitor" alarm will alert the operator if the reading exceeds the setpoint of 100 kcpm.

    c) If the fission product gas release is caused by high temperature of the fissile materials/barriers, an over-temperature alarm will automatically scram the reactor and thus minimize <u>all</u> fission products release.

2. An analysis is performed assuming that, as a result of multiple failures, all fission product gases produced from a fissile materials irradiation experiment are released through the reactor ventilation system. The fission product gases analyzed here are the noble gas nuclides including Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, and Xe-135. The fission product gas inventory is assumed to be at equilibrium at 100 kW (maximum allowable power of a fissile materials irradiation experiment) and is released within one week. The analysis concludes that

    a) The core purge monitor should detect a higher reading, 44 kcpm over background, if the fission product gases were to escape the barriers. The increase is approximately twice that of normal background.

b) If the entire fission product gas inventory leaks from a fissile materials irradiation experiment and is released to the atmosphere through the stack, the total inhalation dose is calculated to be about 17 mrem. There will be no additional thyroid dose because none of the fission product gases affects the thyroid.

Details of the above calculations can be found in the attached MathCAD worksheet.

Fission Product Gas Releases From Fissile Materials Irradiation Experiment

(0: Kr85m, 1: Kr87, 2: Kr88, 3: Xe131m, 4: Xe133m, 5: Xe133, 6: Xe135)

$\mathrm{Ratio} := \frac{100}{6000}$ **(The fission product gas activities are proportional to power generation, fission product inventory obtained from calculations for MITR-III at 6 MW)**

$Ag_0 := 0\,7788 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad Ag_1 := 1.404 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad Ag_2 := 1.92 \cdot 10^5 \cdot \mathrm{Ratio}$

$Ag_3 := 0\,0156 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad Ag_4 := 0.083 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad Ag_5 := 3.37 \cdot 10^5 \cdot \mathrm{Ratio}$

$Ag_6 := 0.50 \cdot 10^5 \cdot \mathrm{Ratio}$

$Ag_0 = 1.298 \cdot 10^3$ $\quad Ag_1 = 2\,34 \cdot 10^3$ $\quad Ag_2 = 3.2 \cdot 10^3$ $\quad Ag_3 = 26$

$Ag_4 = 138.333$ $\quad Ag_5 = 5.617 \cdot 10^3$ $\quad Ag_6 = 833.333$

$\mathrm{Alimit}_0 := 1 \cdot 10^{-7}$ $\quad \mathrm{Alimit}_1 := 2 \cdot 10^{-8}$ $\quad \mathrm{Alimit}_2 := 9 \cdot 10^{-9}$ $\quad \mathrm{Alimit}_3 := 2 \cdot 10^{-6}$

$\mathrm{Alimit}_4 := 6 \cdot 10^{-7}$ $\quad \mathrm{Alimit}_5 := 5 \cdot 10^{-7}$ $\quad \mathrm{Alimit}_6 := 7 \cdot 10^{-8}$

$\mathrm{ALIg}_0 := 0\,7788 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad \mathrm{ALIg}_1 := 1.404 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad \mathrm{ALIg}_2 := 1.92 \cdot 10^5 \cdot \mathrm{Ratio}$

$\mathrm{ALIg}_3 := 0.0156 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad \mathrm{ALIg}_4 := 0\,083 \cdot 10^5 \cdot \mathrm{Ratio}$ $\quad \mathrm{ALIg}_5 := 3.37 \cdot 10^5 \cdot \mathrm{Ratio}$

$\mathrm{ALIg}_6 := 0.50 \cdot 10^5 \cdot \mathrm{Ratio}$

$$\sum_{n=0}^{6} Ag_n = 1.345 \cdot 10^4$$

$i := 0 .. 6$

$$\mathrm{Aconc}_i := \frac{Ag_i \cdot 1 \cdot 10^6}{\left[50000 \cdot 4000 \cdot (12 \cdot 2.54)^3\right] \cdot 60 \cdot 24 \cdot 7}$$

(Stack release concentration in microCi/cc for complete FP gas release over the duration of one week)

(Stack air flow rate ~4000 cfm, 50000 is the dilution factor)

$$Ag = \begin{bmatrix} 1.298\cdot10^{3} \\ 2.34\cdot10^{3} \\ 3.2\cdot10^{3} \\ 26 \\ 138.333 \\ 5.617\cdot10^{3} \\ 833.333 \end{bmatrix} \quad Aconc = \begin{bmatrix} 2.274\cdot10^{-8} \\ 4.099\cdot10^{-8} \\ 5.606\cdot10^{-8} \\ 4.554\cdot10^{-10} \\ 2.423\cdot10^{-9} \\ 9.839\cdot10^{-8} \\ 1.46\cdot10^{-8} \end{bmatrix} \quad Alimit = \begin{bmatrix} 1\cdot10^{-7} \\ 2\cdot10^{-8} \\ 9\cdot10^{-9} \\ 2\cdot10^{-6} \\ 6\cdot10^{-7} \\ 5\cdot10^{-7} \\ 7\cdot10^{-8} \end{bmatrix}$$

$\frac{Aconc_i}{Alimit_i} =$	$DAC_i :=$
0.227	$2\cdot10^{-5}$
2.05	$5\cdot10^{-6}$
6.228	$2\cdot10^{-6}$
$.277\cdot10^{-4}$	$4\cdot10^{-4}$
$.039\cdot10^{-3}$	$1\cdot10^{-4}$
0.197	$1\cdot10^{-4}$
0.209	$1\cdot10^{-5}$

$$\sum_{n=0}^{6} Aconc_n = 2.356\cdot10^{-7}$$

$$Dose_i := \frac{24\cdot7}{2000}\cdot5000\cdot\frac{Aconc_i}{DAC_i}$$

(Dose received from inhalation within a week)

(The doses are calculated using inhalation DAC (occupational values) in 10 CFR 20 deverived for 2000 hours per working year)

$$Dose = \begin{bmatrix} 0.477 \\ 3.443 \\ 11.772 \\ 4.782\cdot10^{-4} \\ 0.01 \\ 0.413 \\ 0.613 \end{bmatrix}$$

(The doses for each isotope are in mR)

$$\sum_{n=0}^{6} Dose_n = 16.729$$

$$\text{Purge} := \frac{(100-20)\cdot 1000}{3.7\cdot 10^{10}}\cdot\frac{1}{0.2}\cdot 10^{6}$$

(core purge set point 100k cpm, assuming detector efficiency 20% Purge has the unit of microCi per minute, background normally at 20 kcpm)

$$\text{Purge} = 10.811$$

(core purge activity in microCi per minutes corresponds to the set point)

$$\text{FPgasrate} := \frac{\displaystyle\sum_{n=0}^{6} Ag_n \cdot 10^{6}}{\left[10\cdot(12\cdot 2.54)^{3}\right]\cdot(60\cdot 24\cdot 7)}$$

(Here is an approximation of a constant FP gas activity in microCi per minute over a week in the air space (~ 10 ft^3) of the upper core tank. Normal core purge flow rate is 4 cpm)

$$\text{FPgasrate} = 4.713$$

$$\frac{\text{FPgasrate}}{\text{Purge}} = 0.436$$

(The core purge monitor will be about 44 kcpm higher if fission product gases leak)

Page

7. Radioactive Releases

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

Basis

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

Specifications 2, 3, 4, 5 and 6 are intended to minimize the probability of experiment failures. Experiment capsules should be designed to withstand expected temperatures, pressures, chemical and radiochemical effects. The requirement for testing containers at twice the pressure or with twice the amount of explosive or metastable material to be irradiated provides a factor of 2 safety margin as allowance for experimental uncertainties. Table 6.1-1 gives a summary of the requirements for specimen irradiations for ease of review and classification of the specifications.

The radiological consequences of experiment malfunctions must be considered as stated in Specification 7. Consistent with the Commission's regulations, predicted onsite personnel exposures or offsite concentrations resulting from these malfunctions must not be in excess of those permitted by 10 CFR Part 20.

6.7 Experiments Involving In-Core Irradiation of Fissile Materials

Applicability

This specification applies to the in-core irradiation of fissile materials. It does not apply to out-of-core irradiations.

Objective

To ensure that fissile materials experiments do not affect safe operation of the reactor and to provide for the protection of the public health and safety by ensuring the integrity of irradiated fissile materials.

Specification

1. In-core fissile materials irradiation experiments shall not contain circulating loops.
2. The physical form of the fissile materials shall be solid. The fissile materials shall be doubly encapsulated to preclude radionuclide leakage during irradiation.
3. The cross section of an in-core fissile materials experiment facility shall not exceed 16 square inches.
4. The total initial amount of U-235 in each in-core fissile materials experiment shall not exceed 100 grams. Any mixture of fissile materials is permitted provided that the off-site dose consequences are less than those of 100 grams of U-235.
5. Thermal power generated from each fissile materials experiment shall not exceed 100 kW during irradiation.
6. Each fissile materials irradiation experiment shall be monitored so that over-temperature protection is provided by automatic reactor scram.
7. Any void space between the inner and outer barriers of the double encapsulation shall be sampled at least weekly for indication of fission products during any week that the experiment is in core and the reactor power exceeds 100 kW. The finding shall be compared to a baseline and the reactor power shall be made less than 100 kW if fission product activity exceeds three times baseline.

8. Design of the fissile materials experiments shall conform to the provisions of TS #6.1 "General Experiment Criteria". Each proposed in-core fissile materials experiment shall require a documented safety review and approval by the MIT Reactor Safeguards Committee (MITRSC) or, if authorized by the MITRSC, by its Subcommittee for in-core experiments.

Basis

The MITR is licensed as a research reactor. Code of Federal Regulations 10 Part 50.2 defines a non-power reactor as a research or test reactor licensed under 10 CFR 50.21(c) or 50.22 for research and development. A test facility is defined in 10 CFR 50.2 as a nuclear reactor for which "...an application has been filed for a license authorizing operation at: (1) a thermal power level in excess of 10 megawatts; or (2) a thermal power level in excess of 1 megawatt, if the reactor is to contain: (i) a circulating loop through the core in which the applicant proposes to conduct fuel experiments; or (ii) a liquid fuel loading; or (iii) an experimental facility in the core in excess of 16 square inches in cross-section." Therefore, Technical Specifications 6.7.1, 6.7.2, and 6.7.3 are based on 50.2(2)(i), 50.2(2)(ii), and 50.2(2)(iii), respectively.

Other limits on the in-core irradiation of fissile materials specific to the MITR in-core experiments are experiment reactivity worth limit and onset of nucleate boiling (ONB). Additional requirements such as weekly sampling of cover gas in the void space and over-temperature automatic reactor scram provide redundant protection against a potential malfunction of the fissile materials irradiation experiments. The limit on U-235 content in a fissile materials irradiation experiment is derived from the Design Basis Accident (DBA) of the reactor. The effect of actinides, which are produced from U-238, on off-site dose is analyzed. It is concluded that a limit on the initial amount of U-238 is not required.

The Design Basis Accident (DBA) chosen for the MITR assumes a blockage of five coolant flow channels that results in four fuel plates completely melted [6.7-1]. Release of the fission products to the atmosphere is calculated assuming that the fission product buildup achieved saturation. Off-site dose to the general public is then calculated from the released fission products. The maximum amount of fissile materials that can be accommodated in a fissile materials experiment should result in a maximum fission product release below that of the DBA. Using an approximation based on the U-235 content, the maximum amount of U-235 would be 506 grams (mass of U-235 per fuel element) x 4 (plates) ÷ 15 (plates per fuel element) = 135 grams. A limit of the total initial amount of 100 grams U-235 is conservatively chosen.

Actinides are produced when U-238 is irradiated. The off-site whole body dose from actinides was calculated to be 2 mrem per kilogram of initial U-238 [6.7-2]. The maximum initial amount of U-238, which is set by the total off-site dose from both fission products and actinides releases of the fissile materials experiment, was calculated to be 31 kilograms. This amount is significantly higher than that of natural uranium that contains 100 grams of U-235, 0.1 kgU-235×(0.993/0.007)= 14.2 kgU-238. Therefore, a limit on the initial amount of U-238 is not required.

The limit on the thermal power generated from the fissile materials experiment is primarily imposed by the onset of nucleate boiling (ONB), which is one of the criteria in TS #6.1. Each in-core fissile materials experiment design will be reviewed to ensure that ONB would not occur during steady-state operation. However, 100 kW, which is less than the average thermal power per fuel element of 200 kW, is used to set an upper bound for any fissile materials irradiation experiment.

The inner barrier of the double encapsulation is monitored for over-temperature. The scram setpoint is experiment-dependent and will be chosen to avoid rapid mechanical and/or chemical degradation of the barrier. The scram setpoint will be documented in the safety review that is required by provision 8 of this technical specification.

Fission product gas release is an unlikely accident scenario An analysis is performed assuming that, as a result of multiple failures, the entire fission product gas inventory produced from a fissile materials irradiation experiment is released through the reactor ventilation system. The fission product gases analyzed here are the noble gas nuclides including Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, and Xe-135. The fission product gas inventory is assumed to be at equilibrium at 100 kW (maximum allowable power of a fissile materials irradiation experiment) and is released within one week. The interval of one week is chosen because that is the frequency for the void space sampling. This is a conservative assumption because a shorter interval will result in a much higher core purge monitor reading and hence increase the probability of detection. The analysis concludes that (a) the core purge monitor should detect a higher reading, 44 kcpm over background, if the fission product gases were to escape the barriers, an increase equivalent to approximately twice that of a normal background reading, and (b) if the entire fission product gas inventory leaks from a fissile materials irradiation experiment and is released to the atmosphere through the stack, the total inhalation dose is calculated to be about 17 mrem. The inhalation dose of 17 mrem is much lower than the 100 mrem annual limit for general public defined by 10 CFR 20. There will be no additional thyroid dose because none of the fission product gases affect the thyroid.

References

6.7-1 MITR Staff, "Safety Analysis Report for the MIT Research Reactor (MITR-II)," Report No. MITNE-115, 22 Oct. 1970.

6.7-2 File Memo "Actinides Off-Site Dose Calculations During DBA, " Oct. 2001.

6.7-3 File Memo " Off-Site Dose Calculations for Fission Product Gases Release, " August 2002.