NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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Activation Analysis
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November 21, 2001

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

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"

Gentlemen:

The Massachusetts Institute of Technology hereby submits a request to amend its operating license (R-37) to permit the conduct of experiments involving the in-core irradiation of fissile materials. Enclosed please find:

1. Proposed MITR Technical Specification No. 6.7, "Experiments Involving In-Core Irradiation of Fissile Materials". This is a new specification.

2. A revision of MITR Technical Specification No. 6.1.7(b) which concerns general experiment criteria as relates to radioactive releases.

3. MITR Safety Reviews #0-01-11 and #0-01-12 which support the above amendment changes.

Please contact either the undersigned or Dr. Lin-Wen Hu of my staff should further information be required. This request has been reviewed and approved by the MIT Committee on Reactor Safeguards.

Sincerely,

John A. Bernard, Ph.D.
Director
MIT Nuclear Reactor Laboratory

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

Executed on 11-21-01 _____
Date Signature

JAB/gw

cc: USNRC - Senior Project Manager,
NRR/ONDD
USNRC - Region I - Project Scientist,
Effluents Radiation Protection Section (ERPS)
FRSSB/DRSS

A020

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 material 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 contained 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. Based on the maximum credible reactor accident that consists of melting of 4 fuel plates, 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. 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.

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]. Note that 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 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 material experiment should result in a maximum fission product release below that of the DBA. 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 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 kilograms 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 fission 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 material 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 that of the average thermal power per fuel element of 200 kW, is used to set an upper bound for any fissile materials irradiation experiment.

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.

7. Radioactive Releases

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

b. Each experiment or a group of credibly coupled experiments shall be designed and operated such that the maximum dose in unrestricted areas resulting from a credible release of its radioactivity inventory shall not exceed that of the Design Basis Accident.

Bases

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

Safety Review #-0-01-11

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 are the Design Basis Accident (DBA) off-site dose, experiment reactivity worth limit, and onset of nucleate boiling (ONB).

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% ΔK/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.

4. Proposed Wording for TS #6.7

See attached.

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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:

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

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),
- $\overline{F}_f$ 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:

$$D_{wb,actinides} = R \times D_{thyroid,DBA} \qquad (2)$$

where

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

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

	$\overline{F}_f$	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 $\frac{247\text{mrem} - 184\text{mrem}}{2\text{mrem / kgU238}} = 31.5$ 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.

Initial U-238 = 330 grams

	Grams				Activity(Ci)			
B(MWd/kg)	Pu-238	Pu-239	Pu-240	Pu-241	Pu-238	Pu-239	Pu-240	Pu-241
0	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0	0	0	0
6.5925	2.17E-05	3.04E-01	6.49E-03	5.31E-04	0.000371	0.018799	0.001468	0.054702
10.668	1.49E-04	5.92E-01	2.32E-02	3.12E-03	0.002544	0.036609	0.005248	0.321412
14.6025	4.51E-04	8.63E-01	4.85E-02	9.07E-03	0.007701	0.053367	0.010971	0.934362
18.729	1.02E-03	1.08E+00	7.90E-02	2.08E-02	0.017416	0.066786	0.017871	2.142748
22.747	1.85E-03	1.26E+00	1.13E-01	3.58E-02	0.031588	0.077917	0.025562	3.687998
26.618	2.99E-03	1.43E+00	1.50E-01	5.29E-02	0.051053	0.088429	0.033932	5.449584
30.2925	4.48E-03	1.59E+00	1.87E-01	7.53E-02	0.076495	0.098324	0.042302	7.757158
34.017	6.29E-03	1.71E+00	2.22E-01	1.01E-01	0.1074	0.105744	0.05022	10.40469
37.6045	8.56E-03	1.84E+00	2.60E-01	1.25E-01	0.14616	0.113783	0.058816	12.87709

Note: This calculation is based on an initial U-238 amount of 330 grams

Actinides

$\lambda Pu238 := \frac{0.693}{87.7 \cdot 365 \cdot 24 \cdot 3600}$ **(half life 87.7 years)** $\lambda Pu238 = 2.506 \cdot 10^{-10}$

$\lambda Pu239 := \frac{0.693}{24110 \cdot 365 \cdot 24 \cdot 3600}$ **(half life 24110 years)** $\lambda Pu239 = 9.114 \cdot 10^{-13}$

$\lambda Pu240 := \frac{0.693}{6564 \cdot 365 \cdot 24 \cdot 3600}$ **(half life 6564 years)** $\lambda Pu240 = 3.348 \cdot 10^{-12}$

$\lambda Pu241 := \frac{0.693}{14.35 \cdot 365 \cdot 24 \cdot 3600}$ **(half life 14.35 years)** $\lambda Pu241 = 1.531 \cdot 10^{-9}$

Activities of Pu-238, Pu-239, Pu-240, and Pu-241 curve-fitted from Pavel's burnup data up to 37 MWd/kgiHM

$$A\_Pu238(B) := \left[1.4624 \cdot 10^{-6} - \left(6.6136 \cdot 10^{-6} \cdot B\right) + 2.7376 \cdot 10^{-7} \cdot B^2 + 1.5811 \cdot 10^{-7} \cdot B^3\right] \cdot \frac{6 \cdot 10^{23}}{238} \cdot \lambda Pu238 \cdot \frac{1}{3.7 \cdot 10^{10}}$$

$$A\_Pu239(B) := \left(5.7949 \cdot 10^{-2} + 4.9978 \cdot 10^{-2} \cdot B\right) \cdot \left(\frac{6 \cdot 10^{23}}{239} \cdot \lambda Pu239 \cdot \frac{1}{3.7 \cdot 10^{10}}\right)$$

$$A\_Pu240(B) := \left(-2.6174 \cdot 10^{-2} + 3.7538 \cdot 10^{-3} \cdot B + 1.0428 \cdot 10^{-4} \cdot B^2\right) \cdot \left(\frac{6 \cdot 10^{23}}{240} \cdot \lambda Pu240 \cdot \frac{1}{3.7 \cdot 10^{10}}\right)$$

$$A\_Pu241(B) := \left[5.9919 \cdot 10^{-4} - \left(8.8679 \cdot 10^{-4} \cdot B\right) + 9.9291 \cdot 10^{-5} \cdot B^2 + 3.4978 \cdot 10^{-7} \cdot B^3\right] \cdot \left(\frac{6 \cdot 10^{23}}{241} \cdot \lambda Pu241 \cdot \frac{1}{3.7 \cdot 10^{10}}\right)$$

$BU := 100$

A_Pu238(BU) = 2.735 A_Pu239(BU) = 0.313

A_Pu240(BU) = 0.315 A_Pu241(BU) = 129.275

$i := 0 .. 8$ **(Beta emitters 0: U237, 1: U239, 2: Np238, 3: 239, 4: Pu241, Alpha emitters 5: Pu238, 6: Pu239, 7: Pu240, 8: Cm242)**

$A_0 := 2.41 \cdot 10^2$ $A_1 := 4.60 \cdot 10^3$ $A_2 := 35.6$ $A_3 := 4.58 \cdot 10^3$ $A_4 := A\_Pu241(BU)$

$A_5 := A\_Pu238(BU)$ $A_6 := A\_Pu239(BU)$ $A_7 := A\_Pu240(BU)$ $A_8 := 0.777$

$ALI_0 := 2 \cdot 10^3$ $ALI_1 := 2 \cdot 10^5$ $ALI_2 := 60$ $ALI_3 := 2 \cdot 10^3$ $ALI_4 := 0.3$

$ALI_5 := 7 \cdot 10^{-3}$ $ALI_6 := 6 \cdot 10^{-3}$ $ALI_7 := 6 \cdot 10^{-3}$ $ALI_8 := 0.3$

Iodines (0: I131, 1: I132, 3: I133, 3: I134, 4: I135)

$j := 0 .. 4$

$AI := (1.25 + 1.9 + 2.81 + 3.29 + 2.55)$

$A\_iodine := 7.33 \cdot 10^3$

$AIodine_0 := 1.25 \cdot \frac{A\_iodine}{AI}$ $AIodine_1 := 1.9 \cdot \frac{A\_iodine}{AI}$ $AIodine_2 := 2.81 \cdot \frac{A\_iodine}{AI}$

$AIodine_3 := 3.29 \cdot \frac{A\_iodine}{AI}$ $AIodine_4 := 2.55 \cdot \frac{A\_iodine}{AI}$

$ALI\_Iodine_0 := 2 \cdot 10^2$ $ALI\_Iodine_1 := 1 \cdot 10^4$ $ALI\_Iodine_2 := 9 \cdot 10^2$

$ALI\_Iodine_3 := 5 \cdot 10^4$ $ALI\_Iodine_4 := 4 \cdot 10^3$

Te-132

$$ATe132 := 5.04 \cdot 10^3 \cdot \frac{1.9}{(0.0242 + 0.108 + 0.147 + 0.432 + 0.19 + 1.25 + 1.9 + 1.99 + 2.9)}$$

$ALI\_Te132 := 8 \cdot 10^2$

Retention factors

$Filter\_I := 1$ $Filter\_Ac := 1$ $Filter\_Te := 1$ **(Charcoal Filter efficiency)**

$Depl\_I := 0.3$ $Depl\_Ac := 1$ $Depl\_Te := 1$ **(Depletion within 2 hours)**

$RCS\_I := 0.9$ $RCS\_Ac := 1 \cdot 10^{-4}$ $RCS\_Te := 0.23$ **(Release fraction to RCS)**

$$Ratio\_beta := \frac{\left(\sum_{n=0}^{4} \frac{A_n \cdot 5}{ALI_n} \right) \cdot Filter\_Ac \cdot Depl\_Ac \cdot RCS\_Ac}{\left(\sum_{n=0}^{4} \frac{AIodine_n \cdot 50}{ALI\_Iodine_n} \right) \cdot Filter\_I \cdot Depl\_I \cdot RCS\_I + \frac{ATe132}{ALI\_Te132} \cdot Filter\_Te \cdot Depl\_Te \cdot RCS\_Te}$$

$$\text{Ratio\_alpha} := \frac{\left(\sum_{n=5}^{8} \frac{A_n \cdot 5}{ALI_n}\right) \cdot \text{Filter\_Ac} \cdot \text{Depl\_Ac} \cdot \text{RCS\_Ac}}{\left(\sum_{n=0}^{4} \frac{\text{AIodine}_n \cdot 50}{\text{ALI\_Iodine}_n}\right) \cdot \text{Filter\_I} \cdot \text{Depl\_I} \cdot \text{RCS\_I} + \frac{\text{ATe132}}{\text{ALI\_Te132}} \cdot \text{Filter\_Te} \cdot \text{Depl\_Te} \cdot \text{RCS\_Te}}$$

$\text{Ratio\_beta} = 2.511 \cdot 10^{-3}$ **(Ratio of the whole body dose contributed to the actinides beta emitters to thyroid dose)**

$\text{Ratio\_alpha} = 2.881 \cdot 10^{-3}$ **(Ratio of the whole body dose contributed to the actinides alpha emitters to thyroid dose)**

$\frac{\text{Ratio\_beta}}{330} = 7.61 \cdot 10^{-6}$

(Ratios per gram of U-238)

$\frac{\text{Ratio\_alpha}}{330} = 8.731 \cdot 10^{-6}$

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 material 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 contained 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. Based on the maximum credible reactor accident that consists of melting of 4 fuel plates, 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. 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.

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]. Note that 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 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 material experiment should result in a maximum fission product release below that of the DBA. 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 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 kilograms 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 fission 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 material 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 that of the average thermal power per fuel element of 200 kW, is used to set an upper bound for any fissile materials irradiation experiment.

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.

Safety Review #-0-01-12 – TS #6.1.7 "General Experiment Criteria – Radioactive Releases"

Description of Change

The original wording of this specification was extremely restrictive in that it required a worst-case calculation of any potential release. No allowance was permitted for realistic release pathways.

Safety Evaluation

The proposed change retains the intent of the original wording which is to limit the dose in unrestricted areas. However, it allows estimation of that dose to be calculated using 'credible releases'. That is, actual pathways can be considered.

7. Radioactive Releases

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

b. Each experiment or a group of credibly coupled experiments shall be designed and operated such that the maximum dose in unrestricted areas resulting from a credible release of its radioactivity inventory shall not exceed that of the Design Basis Accident.

Bases

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

7. Radioactive Releases

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

b. 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 (Section 5.3.1 of the SAR). (Replace with new text.)

Bases

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