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22 August 2017

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC

Re: Technical Specifications Amendment
License No. R-120
Docket No. 50-297

Attached please find an amendment request regarding Technical Specification 3.8 for fueled experiments. This request is based on a revised analysis and replaces the previous submittal made on 12 January 2017.

If you have any questions regarding this amendment or require additional information, please contact Gerald Wicks at 919-515-4601 or wicks@ncsu.edu .

I declare under penalty of perjury that the forgoing is true and correct. Executed on 22 August 2017.

Sincerely,

Ayman I. Hawari

Ayman I. Hawari, Ph.D.
Director, Nuclear Reactor Program
North Carolina State University

Enclosures: Technical Specification Amendment
Attachment 1: Fueled Experiment Analysis

cc: Duane Hardesty, US NRC

Summary

An amendment to the Technical Specifications (TS) is requested regarding TS 3.8 for fueled experiments. TS 3.8 has been modified for conducting experiments using fissionable materials based on a limiting fission rate as analyzed in Attachment 1 for U-235 and several other fissionable materials. The limiting fission rate limit allows use of any fissionable material in fueled experiments that would not exceed annual radiation dose limits to occupational workers or the public from a credible accident.

Attachment 1 provides details on the use, production, and release of radioactive material from U-235 and several other fissionable materials. At the limiting fission rate, activities of fission product gases and halogens are at saturation levels. Fission products of concern for a release are isotopes of Kr, Xe, I, and Br. Cs and Rb particulates from the decay of released Xe and Kr are also considered. Release of particulates directly from the sample container are not considered since only non-respirable particles are allowed and release of particulates is reported to be several orders of magnitude lower than the release of noble gases and halogens. All fission gases and halogens are assumed to be released. No decay corrections are made post-production prior to release or during transport by the atmosphere following the release since a failure may occur anytime during the experiment.

The released activity into the reactor building air space and subsequent venting to the environment was analyzed for radiation dose from inhalation and submersion dose pathways to occupants inside the reactor building and in public areas outside the reactor building. Credit is taken for protective and mitigating actions to limit radiation dose, e.g. sample encapsulation, radiation monitoring, evacuation of the reactor building, filtration, and atmospheric dispersion of radioactive effluent. Credit for decay inside the reactor building is also taken.

The definition of fueled experiments is changed in TS 1.2.9 e to provide a maximum fission rate that does not exceed the annual public dose limit and to exclude self-encapsulated items such as detectors, foils, wire, sealed sources, and fuel with cladding.

TS 3.8 limits individual fueled experiments. TS 3.8 as proposed meets 10 CFR Part 20, Category 2 limits given in 10 CFR Part 37, other TS requirements on experiments, and the facility emergency plan and security plan. Conditions do not exceed those for a reportable event, or emergency action level.

Possession limits in the reactor license are changed to specifically identify limits for fueled experiments and to remain below 10 CFR Part 37 Category 2 limits. As a result of not exceeding 10 CFR Part 37 Category 2 limits, the facility security plan is not revised. Materials are stored as required by TS and the facility security plan and radiation protection program.

R-120 License Change:

2 B. (2) Pursuant to the Act and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material,” to receive, possess, and use in connection with operation of the reactor up to 25 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel; up to 20 grams of contained uranium-235 of any enrichment in the form of fission chambers; up to 2 grams of contained uranium-235 of any enrichment in the form of foils; up to 200 grams of plutonium-239 in the form of plutonium-beryllium neutron sources; as listed in the table below for fissionable materials in non-volatile form of any enrichment for fueled experiments; and to possess, but not separate, such special nuclear material as may be produced by operation of the facility or by performance of fueled experiments.

FUELED EXPERIMENT POSSESSION LIMITS

Fissionable Material	Mass Limit (g)	Sum of Fractions
Pu238 Pu239 Am241 Cm244 Cf252	0.47 100 2.2 0.08 0.005	Not to exceed 0.9 for all
Special Nuclear Material not listed above	100 each or 200 total	

TS Changes:

1.2.9 e.

Fueled Experiment is an experiment that involves the use of fissionable material within the reactor building which is exposed to a neutron fluence, is not self-encapsulated, and has a fission rate greater than 3.7×10^{10} fissions per second.

Fueled experiments exclude:

(i) Fissionable materials not subjected to a neutron fluence

(ii) Self-encapsulated materials, such as detectors and foils and wires, sealed sources, and fuel with cladding

(iii) Fissionable material used in experiments subjected to a neutron fluence with a total fission rate that does not exceed 3.7×10^{10} fissions per second

3.8 Operations with Fueled Experiments

Applicability

This specification applies to the operation of the reactor with any fueled experiment.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

Fueled experiments shall meet the following conditions and limitations:

a. Physical forms of fissionable materials shall be limited to:

   i. Non-volatile materials in the form of liquid, powder, and/or solid. Volatile or gaseous forms of materials are prohibited.

   ii. Particle sizes greater than 10 microns for solid materials.

b. The reactor shall not be operated with a fueled experiment unless the ventilation system is operated in the confinement mode.

c. Total fission rate shall not exceed 1.2×10^{12} fissions per second

d. Specifications 3.2, 3.5, 3.6, and 3.7 pertaining to limiting conditions of operation shall be met.

e. Specification 5.3 pertaining to storage shall be met.

f. Specifications 6.2.3 and 6.5 pertaining to the review of experiments shall be met. Each type of fueled experiment shall be classified as a new (untried) experiment with a documented review. The documented review shall include the following items:

   i. Meeting license requirements for the receipt, use, storage, and security of fissionable material.

   ii. Meeting conditions given in Specification 3.8 a through 3.8 e

   iii. Identification and amounts of fissionable materials present initially and over the planned duration of the fueled experiment

   iv. Production and depletion of fissionable materials over the planned duration of the fueled experiment

v. Highest fission rate expected over the planned duration of the fueled experiment

vi. Irradiation and unloading of irradiated fueled experiments within the reactor building

vii. Initiation of the reactor building evacuation alarm.

Bases

The limitations given in Specification 3.8 ensure that (1) fueled experiments performed in experimental facilities at the reactor prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure and (2) radiation doses to occupational personnel and the public do not exceed radiation dose limits given in 10 CFR Part 20.

Specification 3.8 f ensures that each type of fueled experiment is reviewed, approved, and documented as required by Specifications 6.2.3 and 6.5. The required review content given in Specification 3.8 f ensures safety requirements are met for a fueled experiment.

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ATTACHMENT 1

FUELED EXPERIMENT ANALYSIS

TECHNICAL SPECIFICATIONS AMENDMENT

22 AUGUST 2017

INTRODUCTION

An amendment to the Technical Specifications (TS) is requested regarding TS 3.8 for fueled experiments. TS 3.8 is revised to provide limiting conditions on fission rate for any fissionable material individually or in mixtures. By limiting the fission rate, the radioactive material inventory of fission products saturate and, if released, are calculated not to exceed annual radiation dose limits to occupational workers or the public.

Limitations for a fueled experiment are based on potential radiation dose and other experimental limitations, e.g. reactivity, heat, pressure, materials, and encapsulation. Due to the production of gases and vapors, fueled experiments are to be doubly encapsulated as required by TS 3.7. This analysis is concerned only with potential radiation doses to occupants inside the reactor building and members of the public outside the reactor building following an accidental release of fission products from an encapsulation failure.

While encapsulated, only radiation dose from the contained irradiated sample occurs and is controlled to meet 10 CFR Part 20 annual occupational dose limits by following the facility radiation protection program and other experiment controls as specified by the university review committees. Shielding, limitation of access, and other controls are typically used.

TS 3.8 are revised to state that the sample is limited to non-volatile materials. If in solid form, particle sizes are limited to those greater than 10 microns thereby resulting in negligible release of respirable particulates. Gaseous fissionable materials are prohibited.

A sufficiently long production period is assumed to reach saturate activity of the fission gases and halogens. These include isotopes of Kr, Xe, I, and Br. After being released into the reactor building, Rb and Cs in-growth from the decay of Kr and Xe isotopes are also included. Saturation activity was calculated using a reference conditions for U-235 and several other fissionable materials. Reference conditions were a mass of 1 gram and maximum thermal and non-thermal experiment neutron fluence rates. Cross-section and cumulative fission yield data for U-235 and the maximum value for other fissionable materials were used.

The radioactive material inventory released is assumed to be instantaneously and uniformly distributed throughout the entire reactor bay air space initially and then exhausted to the environment by the reactor building ventilation system which includes filters and an elevated exhaust stack. Concentration inside the reactor building, filter retention, exhaust ventilation rate, and atmospheric dispersion are considered in the analysis. Credit for decay inside the reactor building is also considered. No additional decay corrections are made, e.g. during transport by the atmosphere following the release or decay post-production prior to release since a failure may occur anytime during the experiment. Fission products and fissionable materials in the form of particulates in the irradiated material are not analyzed since particles would be released in insignificant amounts or not at all.

Time integrated exposure and Dose Conversion Factors (DCF) are used to calculate radiation dose. Radiation dose calculated includes; (1) Total Effective Dose-Equivalent (TEDE) and (2) Total Organ Dose-Equivalent (TODE) to the thyroid for occupants inside the reactor building and (3) TEDE for members of the public outside the reactor building. Committed Effective Dose-Equivalent (CEDE) and Deep Dose-Equivalent (DDE) are summed to give the TEDE. Committed Dose-Equivalent (CDE) and DDE are summed to give the TODE to the thyroid. TEDE and TODE to occupants inside the reactor building are limited to 5 rem and 50 rem, respectively. TEDE for members of the public is limited to 0.1 rem. DDE from submersion inside the reactor building is adjusted based on the experimental area dimensions for the reactor building. No adjustments to DDE are made outside the reactor building.

All radiation doses from the release were calculated using saturation activities for U-235 and several other fissionable materials under reference conditions. Reference conditions include a constant mass of 1 gram irradiated at the maximum fluence rates for thermal and non-thermal fission. The worst case conditions used for other fissionable materials were for the highest fission cross-sections and highest cumulative fission yields.

The reference doses for U-235 and for the worse case conditions provide radiation doses per unit fission rate. The reference doses were used to determine the most restrictive, or limiting, fission rate that meets the annual occupational and public radiation dose limits. For worse case conditions, the limiting fission rate is suitable for all fissionable materials analyzed. Given the number of fissionable materials included, this limiting fission rate is applied to any fissionable material.

TS 3.8 specifies the calculated limiting fission rate since it is based on meeting the radiation dose limits. Therefore, the mass and fluence rates in an experiment may be adjusted as needed to meet the specified limiting fission rate.

TS 3.8 limits individual fueled experiments. TS 3.8 as proposed meets 10 CFR Part 20, , Category 2 limits given in 10 CFR Part 37, other TS requirements on experiments, and the facility emergency plan and security plan. Conditions do not exceed those for a reportable event, or emergency action level.

The definition of fueled experiments in TS 1.2.9 e specifies a maximum fission rate such that annual public radiation dose limit (0.1 rem TEDE) is not exceeded either inside or outside the reactor building. Credit is taken for reactor building evacuation. Other exclusions for fueled experiments include experiments using fissionable material that is not exposed to a neutron fluence and self-encapsulated items that would not result in a release of fission products, e.g. wire, foils, sealed sources, and fuel with cladding.

Possession limits in the reactor license are changed to specifically identify limits for fueled experiments and to remain below 10 CFR Part 37 Category 2 limits. As a result of not exceeding 10 CFR Part 37 Category 2 limits, the facility security plan is not revised.

ASSUMPTIONS

Assumed conditions for fueled experiments are as follows:

1. Radioactive materials are encapsulated until the time of failure. Release of radioactive materials from an encapsulation failure are based on ANSI/ANS 15.7 [Ref 1], NUREG 1400 [Ref 2], and US NRC Regulatory Guide 2.2 [Ref 3].

2. Materials are limited to non-volatile liquids and solids. Particle sizes shall be non-respirable, i.e. greater than 10 microns in diameter [Ref 4].

3. Single-mode nonviolent failure of the encapsulation results in release of radioactive noble gases and halogens only into the minimum reactor building free air volume [Ref 3]. Particulate release is not considered due to the materials being non-volatile and the particle sizes being non-respirable.

4. Neutron fluence rate is constant over time and for the entire mass of the fissionable material present during the experiment irradiation time. No correction to the mass is made as a result of activation and fission reactions during the irradiation time.

5. Reactor ventilation system is in the confinement mode [Ref 5]

6. Radioactive noble gases and halogens are assumed to be present at the saturation activity from irradiation at the maximum fluence rate measured in reactor experimental facilities. For thermal neutrons the maximum fluence rate is 7 E12 $cm^{-2}s^{-1}$. For non-thermal neutrons the maximum fluence rate is 2 E12 $cm^{-2}s^{-1}$. [Ref 6]

7. Exposure times to personnel in the reactor building is 3 minutes based on measured and estimated evacuation time from the reactor building.

8. Exposure times to the public is 24 hours based on reactor building exhaust rates in the confinement mode [Ref 7, 8].

9. No credit for respiratory protection is assumed.

10. The release is assumed to occur instantaneously and to be well mixed within the reactor building air space [Ref 1,3]

11. A correction factor of 0.1 is used for submersion dose within the reactor building for photons emitted by noble gases based on dimensions and geometry. A sphere rather than hemisphere is assumed. [Ref 9,10,11]

12. The minimum reactor building free air volume is approximated at 2.25 E9 ml based on reported and measured data. [Ref 5,12]

13. Confinement filter removal efficiency, or retention, is 99.97% for particulates and 99% for halogens [Ref 5]. For conservatism, 90% is the assumed charcoal retention for halogens.

14. Atmospheric dispersion parameter, X/Q at 1 m/s was evaluated for locations from 100 m to 5000 m from the reactor stack, for all weather stability classes and release times of 2 hours and 24 hours. Correction for effective stack height was evaluated, but not used since the difference from the actual stack height was minimal. The most restrictive X/Q value calculated was used to assess public radiation dose. [Ref 1,5]

NOTES

Notes on references and the reactor facility applicable to fueled experiments are given below:

- Chapter 5 of the SAR describes the confinement system flow rate and filters. High Efficiency Particulate Absorber (HEPA) filter retention of 99.97% is stated. Confinement flow rate is 600 cfm. Charcoal filters are mentioned, but no retention is given. Confinement would be placed into operation manually for fueled experiments to meet TS 3.8. Operating procedures include this requirement.

- Chapter 13 of the SAR states the HEPA and charcoal retention are 99.97% and 99% respectively (section 13.2.1.4). In the amendment, 90% is claimed for the charcoal retention for conservatism.

- For operation and fuel movement, the ventilation system is required under TS 3.6 in either normal or confinement mode. If normal mode is used, then the confinement system must be operable - confinement automatically starts for certain conditions or by manual initiation. For fueled experiments, TS 3.8 requires the confinement system to be operated during the experiment.

- Testing is performed per TS 4.5 on the ventilation system, including filter testing in accordance with TS 4.5 e. Maintenance and surveillance is in place for testing of the ventilation system. Acceptance criteria are 99.97% for HEPA tested and 99% for charcoal tested. Testing methods follow ASME N510-1989 "Testing of Nuclear Air Treatment Systems”. Testing is also required following major maintenance of the filters or housing. Testing and maintenance are documented in facility surveillance files as required by TS 6.4 and 6.8.

- Normal ventilation exhaust rate is 1870 cfm and is not filtered prior to release to the environment. Normal ventilation stops and confinement ventilation starts if set points are exceeded or if changed manually. Normal ventilation is the standard ventilation mode for reactor operations.

- As a result of the need for filtered ventilation, an exhaust stack, and radiation monitoring of the ventilation system, fueled experiments are only performed in experimental facilities located inside the reactor building.

- Radiation monitoring of the reactor building and exhaust stack is required for reactor operations under TS 3.5. These include detectors for external radiation levels, radioactive gases, and radioactive particulates. Set points for these monitors meet TS 3.5. For fueled experiments, monitoring for released activity is required as part of the experiment review in the proposed TS 3.8. Set points may be adjusted as needed for experiments and other conditions in accordance with facility procedures. Radiation monitoring is required for reactor operation for both normal or confinement ventilation modes.

- To meet reactor operation requirements, the confinement system and radiation monitoring system must be operable. This includes satisfactory surveillance and testing of the filters, radiation monitors, and building differential pressure (dP).

- Reactor systems, structures, components, plans, and procedures that are in place for fuel handling accidents are applicable for handling of a failed fueled experiment.

SYMBOLS and ABBREVIATIONS

A	atomic mass number or atomic weight in grams per mole
A	radioactivity, where units are specified in the analysis
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
b	cross section in barns
B	branching radioactive decay factor
B	gamma photon buildup factor
β or B	beta particle
C	concentration, units are specified in the analysis
CDE	Committed Dose-Equivalent
CEDE	Committed Effective Dose-Equivalent
D	dispersal factor
$D_{reference}$	Reference dose
DAC	Derived Air Concentration
DCF	Dose Conversion Factor
DDE	Deep Dose-Equivalent
EC	Effluent Concentration
f	submersion gamma photon dose correction factor
F	flow rate
f/s	fission rate
HEPA	High Efficiency Particulate Absorber
k	total removal rate constant
M	mass, units are specified in the analysis
N	number of atoms
P	production rate
$T_{1/2}$	radiological (physical) half-life
T	evacuation time or public exposure time
TEDE	Total Effective Dose-Equivalent
TODE	Total Organ Dose-Equivalent
v	ventilation removal rate constant
V	volume
X/Q	atmospheric dispersion parameter
σ	microscropic (atomic) cross-section
λ	radiological decay constant
Ω	Time integrated exposure

PRODUCTION AND DECAY KINETICS

By assumption, the fission product inventory for the radionuclides available for release attains saturation activity. Saturation activity is estimated as follows [Ref 13, 29]:

$$A(\infty) = k\sigma\varphi NY \qquad \text{EQ 1}$$

Where, $A(\infty)$ is the saturation activity from thermal and non-thermal fission
k is a group conversion constant to give activity
σ is the fission reaction cross section in barns
φ is the neutron fluence rate in $cm^{-2}s^{-1}$
N is the number of atoms for the fissionable material present
Y is the cumulative fission yield for a given radionuclide

For activity in uCi:
$k = (1E\text{-}24\ cm^2/barn)(1\ decay/atom)(1\ uCi\ /3.7E4\ dps) = 2.703E\text{-}29$
N = (Mass in grams)(6.022E23 atoms/mole)(1 mole / atomic mass number)

The maximum fluence rate measured in reactor experimental facilities was used to calculate saturation activity. Saturation activities were calculated and summed for irradiation by thermal and non-thermal fluence rates using reported cross-sections and cumulative fission yields.

Maximum neutron fluence rates:
- For thermal neutrons the maximum fluence rate is 7 E12 $cm^{-2}s^{-1}$ [Ref 6]
- For non-thermal neutrons the maximum fluence rate is 2 E12 $cm^{-2}s^{-1}$ [Ref 6]

Thermal and non-thermal activity calculations from EQ 1 were analyzed separately and together (i.e. summed) since an experiment may be configured for thermal neutrons only, non-thermal neutrons only, or both thermal and non-thermal neutrons.

DECAY DATA

Decay data, e.g. half-lives, were taken from data given in Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Joint Evaluated Fission and Fusion Project Report 20 (JEFF 3.1-3.1.1 Radioactive Decay Data and Fission Yield Sub-Library) [Ref 15].

FLUENCE RATE

Neutron fluence rates in experimental facilities is measured by the reactor staff and shared with the experimenter in planning an experiment. These measurements are made following standard ASTM E261 “Standard Methods for Determining Neutron Fluence, Fluence Rate, and and Spectra by Radioactivation Techniques” [Ref 32] and NIST traceable materials. These measurements are made periodically, as experimental facilities change, or for specific experimental needs.

CUMULATIVE FISSION YIELD DATA

Cumulative fission yields were taken from data given in the following references:

- "Evaluation and Compilation of Fission Product Yields, T.R. England and B.F. Rider, Los Alamos National Laboratory, October, 1994, LA-UR 94-3106 ENDF 349" [Ref 14]
- Japan Atomic Energy Agency Nuclear Data Center Tables of Nuclear Data (JENDL data) [Ref 18].

	Fission Yiled %			
	Max	Max	U-235	U-235
Nuclide	Thermal	Non-thermal	Thermal	Non-thermal
83mKr	5.59E+00	2.22E+00	5.36E-01	5.75E-01
85mKr	5.63E+00	3.47E+00	1.29E+00	1.36E+00
85Kr	5.71E+00	3.47E+00	2.83E-01	2.96E-01
87Kr	7.18E+00	6.95E+00	2.56E+00	2.54E+00
88Kr	7.81E+00	6.88E+00	3.55E+00	3.43E+00
89Kr	8.66E+00	7.47E+00	4.51E+00	3.97E+00
90Kr	6.37E+00	7.96E+00	4.86E+00	4.60E+00
91Kr	3.54E+00	6.25E+00	3.35E+00	3.03E+00
92Kr	1.69E+00	4.12E+00	1.67E+00	1.39E+00
93Kr	5.23E-01	1.87E+00	4.89E-01	3.47E-01
94Kr	1.95E-01	7.34E-01	8.70E-02	6.18E-02
95Kr	3.71E-02	1.27E-01	7.19E-03	2.26E-02
96Kr	3.78E-02	1.09E-02	3.78E-02	2.30E-03
97Kr	1.61E-03	9.31E-04	2.97E-05	1.15E-04
Xe131m	4.21E-02	4.54E-02	4.05E-02	3.54E-02
Xe133m	1.99E-01	1.97E-01	1.89E-01	1.97E-01
Xe133	7.02E+00	6.97E+00	6.70E+00	6.71E+00
Xe135m	1.22E+00	1.15E+00	1.10E+00	1.26E+00
Xe135	7.70E+00	7.54E+00	6.54E+00	6.58E+00
Xe137	6.84E+00	6.58E+00	6.13E+00	5.98E+00
Xe138	7.19E+00	6.27E+00	6.30E+00	6.00E+00
Xe139	6.93E+00	6.90E+00	5.04E+00	4.22E+00
Xe140	4.22E+00	6.77E+00	3.65E+00	2.69E+00
Xe141	2.11E+00	4.41E+00	1.25E+00	8.92E-01
Xe142	8.27E-01	2.29E+00	4.39E-01	5.68E-01
Xe143	1.94E-01	7.03E-01	5.30E-02	1.67E-02
Xe144	3.14E-02	2.34E-01	6.05E-03	2.15E-02
Xe145	2.57E-03	1.98E-02	7.16E-05	8.49E-04
I131	3.86E+00	4.17E+00	2.89E+00	3.22E+00
I132	5.41E+00	5.33E+00	4.31E+00	4.66E+00
I133	7.02E+00	6.97E+00	6.70E+00	6.70E+00
I134	7.82E+00	7.98E+00	7.83E+00	7.63E+00
I135	7.42E+00	6.99E+00	6.28E+00	6.27E+00
83Br	5.59E+00	2.22E+00	5.40E-01	5.76E-01
84Br	9.20E+00	4.09E+00	9.67E-01	1.01E+00
84mBr	3.13E-01	4.19E-01	2.27E-02	1.78E-02
85Br	5.61E+00	3.47E+00	1.30E+00	1.36E+00
86Br	6.01E+00	5.89E+00	1.82E+00	1.73E+00
87Br	4.78E+00	6.80E+00	2.07E+00	2.11E+00
88Br	2.72E+00	6.03E+00	1.74E+00	2.12E+00
89Br	1.31E+00	5.02E+00	1.08E+00	1.44E+00
90Br	5.53E-01	2.62E+00	5.64E-01	7.47E-01
91Br	2.25E-01	1.01E+00	2.24E-01	1.37E-01
92Br	3.83E-02	2.34E-01	2.68E-02	2.09E-02
93Br	8.31E-03	6.42E-02	3.08E-03	6.41E-02
96Br	1.91E-06	3.52E-05	1.91E-06	2.28E-07

CROSS-SECTION DATA

Fission cross-section data for thermal and non-thermal neutron energies used in this analysis were evaluated at:

- 0.025 eV for thermal neutron energy
- For non-thermal energies, the higher of the following was used
  - Average from 1E-5 eV to 10 eV
  - Resonance integral from 0.5 to 1E5 eV

References reviewed for fission cross section data included:

- National Nuclear Data Center, Brookhaven National Laboratory, Evaluated Nuclear Data Files (ENDF libraries) [Ref 16]
- OECD NEA Joint Evaluated Fission and Fusion Project Report 21 [Ref 17]
- Japan Atomic Energy Agency Nuclear Data Center Tables of Nuclear Data (JENDL data) [Ref 18]

To evaluate the limiting fission rate for U-235 and other fissionable materials, the thermal fission yields and non-thermal cumulative fission yields and fission cross-sections for the following were reviewed:

Thermal fission:	Th227, Th229 U232, U233, U235 Np237 Pu239, Pu240, Pu242 Am241, Am242 Cm245 Cf249, Cf251 Fm255
Non-thermal fission:	Pa231 U233, U234, U235, U236, U237, U238 Np237, Np238 Pu238, Pu239, Pu240, Pu241, Pu242 Am241, Am243 Cm242, Cm243, Cm244. Cm246, Cm248

Cross-Section Data:

Nuclide	Thermal Fission Cross Section (b)	Non-thermal Fission Cross section (b)
Pa-231	1.04E-02	9.82E-01
Th-227	2.02E+02	2.06E+02
Th-229	3.08E+01	3.14E+01
Th233	1.50E+01	1.51E+01
U-232	7.71E+01	7.53E+01
U-233	5.31E+02	5.29E+02
U-234	6.70E-02	6.63E-02
U-235	5.85E+02	5.71E+02
U-236	6.13E-02	6.14E-02
U-237	1.70E+00	1.67E+00
U-238	2.65E-05	2.65E-05
Np-237	2.04E-02	1.99E-02
Np-238	2.03E+03	2.01E+03
Pu-238	1.79E+01	1.71E+01
Pu-239	7.48E+02	7.89E+02
Pu-240	5.92E-02	6.11E-02
Pu-241	1.01E+03	1.06E+03
Pu-242	2.56E-03	2.57E-03
Am-241	3.15E+00	3.33E+00
Am-242	2.09E+03	2.20E+03
Am-243	8.13E-02	8.24E-02
Cm-242	5.06E+00	5.04E+00
Cm-243	6.18E+02	6.20E+02
Cm-244	6.04E-01	5.97E-01
Cm-245	2.14E+03	2.03E+03
Cm-246	1.44E-01	1.45E-01
Cm-248	3.70E-01	3.70E-01
Cf-249	1.67E+03	1.61E+03
Cf-251	5.32E+03	5.31E+03
Fm-255	3.36E+03	3.36E+03
Max=	5.32E+03	5.31E+03

For U-235, the fission cross-sections used were 585 barns for thermal neutrons and 571 barns for non-thermal neutrons.

The maximum fission cross-sections were for Cf-251 at 5320 barms for thermal neutrons and 5310 barns for non-thermal neutrons.

SATURATION ACTIVITY

U-235 saturation activities were calculated and compared to those using Nuclear Analysis 1.0 [Ref 19] for a mass of 1 g of U-235 irradiated by a thermal fluence rate of 7 E12 $cm^{-2}s^{-1}$:

Nuclide	Cumulative Yield per 100 fissions	Calculated Activity, uCi	Nuclear Analysis uCi
83mKr	5.36E-01	1.52E+06	
85mKr	1.29E+00	3.66E+06	
85Kr	2.83E-01	8.03E+05	5.30E+05
87Kr	2.56E+00	7.26E+06	7.10E+06
88Kr	3.55E+00	1.01E+07	9.55E+06
89Kr	4.51E+00	1.28E+07	1.28E+07
90Kr	4.86E+00	1.38E+07	1.40E+07
91Kr	3.35E+00	9.50E+06	9.58E+06
92Kr	1.67E+00	4.74E+06	4.78E+06
93Kr	4.89E-01	1.39E+06	1.39E+06
94Kr	8.70E-02	2.47E+05	2.37E+05
95Kr	7.19E-03	2.04E+04	1.96E+04
96Kr	3.78E-02	1.07E+05	1.03E+05
97Kr	2.97E-05	8.42E+01	8.10E+01
131mXe	4.05E-02	1.15E+05	
133mXe	1.89E-01	5.36E+05	
133Xe	6.70E+00	1.90E+07	1.01E+07
135mXe	1.10E+00	3.12E+06	
135Xe	6.54E+00	1.85E+07	9.70E+06
137Xe	6.13E+00	1.74E+07	1.75E+07
138Xe	6.30E+00	1.79E+07	1.79E+07
139Xe	5.04E+00	1.43E+07	1.44E+07
140Xe	3.65E+00	1.04E+07	1.03E+07
141Xe	1.25E+00	3.55E+06	3.56E+06
142Xe	4.39E-01	1.25E+06	1.25E+06
143Xe	5.30E-02	1.50E+05	1.50E+05
144Xe	6.05E-03	1.72E+04	1.70E+04
145Xe	7.16E-05	2.03E+02	2.02E+02
131I	2.89E+00	8.20E+06	7.20E+06
132I	4.31E+00	1.22E+07	9.40E+06
133I	6.70E+00	1.90E+07	1.00E+07
134I	7.83E+00	2.22E+07	2.10E+07
135I	6.28E+00	1.78E+07	1.77E+07
83Br	5.40E-01	1.53E+06	1.40E+06
84Br	9.67E-01	2.74E+06	2.40E+06
84mBr	2.27E-02	6.43E+04	
85Br	1.30E+00	3.68E+06	2.27E+06
86Br	1.82E+00	5.15E+06	3.08E+06
87Br	2.07E+00	5.87E+06	5.82E+06
88Br	1.74E+00	4.93E+06	4.70E+06
89Br	1.08E+00	3.07E+06	3.10E+06
90Br	5.64E-01	1.60E+06	1.57E+06
91Br	2.24E-01	6.35E+05	6.36E+05
92Br	2.68E-02	7.59E+04	7.60E+04
93Br	3.08E-03	8.74E+03	8.75E+03
96Br	1.91E-06	5.41E+00	

Some radionuclides in Nuclear Analysis 1.0 have no fission yield data (e.g. Kr-83m). Also, minor differences between the references used in the calculation and Nuclear Analysis 1.0 data libraries are associated with fission cross-sections and cumulative fission yields. The library data used in the analysis is more recent than that used in Nuclear Analysis 1.0.

In general, there is good agreement for most radionuclides. For other radionuclides, the calculation used in this analysis gave higher saturation activities.

Saturation activities calculated for a mass of 1 g of U-235 irradiated by a thermal fluence rate of 7 E12 $cm^{-2}s^{-1}$ and a non-thermal fluence rate of 2 E12 $cm^{-2}s^{-1}$ are as follows:

Nuclide	Half-Life s	Decay Constant 1/s	Cumulative Yield % Thermal Fission	Cumulative Yield % Non- Thermal Fission	Calculated Activity, uCi
83mKr	6.70E+03	1.04E-04	***5.36E-01***	***5.75E-01***	1.97E+06
85mKr	1.61E+04	4.30E-05	***1.29E+00***	***1.36E+00***	4.73E+06
85Kr	3.39E+08	2.05E-09	***2.83E-01***	***2.96E-01***	1.04E+06
87Kr	4.57E+03	1.52E-04	***2.56E+00***	***2.54E+00***	9.27E+06
88Kr	1.02E+04	6.78E-05	***3.55E+00***	***3.43E+00***	1.28E+07
89Kr	1.89E+02	3.67E-03	***4.51E+00***	***3.97E+00***	1.59E+07
90Kr	3.23E+01	2.15E-02	***4.86E+00***	***4.60E+00***	1.74E+07
91Kr	8.60E+00	8.06E-02	***3.35E+00***	***3.03E+00***	1.19E+07
92Kr	1.84E+00	3.77E-01	***1.67E+00***	***1.39E+00***	5.83E+06
93Kr	1.29E+00	5.37E-01	***4.89E-01***	***3.47E-01***	1.66E+06
94Kr	2.10E-01	3.30E+00	***8.70E-02***	***6.18E-02***	2.96E+05
95Kr	7.80E-01	8.89E-01	***7.19E-03***	***2.26E-02***	3.83E+04
96Kr	8.00E-02	8.66E+00	***3.78E-02***	***2.30E-03***	1.09E+05
97Kr	6.30E-02	1.10E+01	***2.97E-05***	***1.15E-04***	1.75E+02
131mXe	1.03E+06	6.74E-07	***4.05E-02***	***3.54E-02***	1.43E+05
133mXe	1.89E+05	3.66E-06	***1.89E-01***	***1.97E-01***	6.92E+05
133Xe	4.53E+05	1.53E-06	***6.70E+00***	***6.71E+00***	2.43E+07
135mXe	9.18E+02	7.55E-04	***1.10E+00***	***1.26E+00***	4.12E+06
135Xe	3.28E+04	2.12E-05	***6.54E+00***	***6.58E+00***	2.38E+07
137Xe	2.29E+02	3.02E-03	***6.13E+00***	***5.98E+00***	2.21E+07
138Xe	8.46E+02	8.19E-04	***6.30E+00***	***6.00E+00***	2.26E+07
139Xe	3.97E+01	1.75E-02	***5.04E+00***	***4.22E+00***	1.76E+07
140Xe	1.36E+01	5.10E-02	***3.65E+00***	***2.69E+00***	1.25E+07
141Xe	1.72E+00	4.03E-01	***1.25E+00***	***8.92E-01***	4.25E+06
142Xe	1.22E+00	5.68E-01	***4.39E-01***	***5.68E-01***	1.69E+06
143Xe	3.00E-01	2.31E+00	***5.30E-02***	***1.67E-02***	1.64E+05
144Xe	1.20E+00	5.78E-01	***6.05E-03***	***2.15E-02***	3.42E+04
145Xe	1.88E-01	3.69E+00	***7.16E-05***	***8.49E-04***	8.75E+02
131I	6.93E+05	1.00E-06	***2.89E+00***	***3.22E+00***	1.07E+07
132I	8.26E+03	8.39E-05	***4.31E+00***	***4.66E+00***	1.59E+07
133I	7.49E+04	9.26E-06	***6.70E+00***	***6.70E+00***	2.43E+07
134I	3.16E+03	2.20E-04	***7.83E+00***	***7.63E+00***	2.82E+07
135I	2.37E+04	2.93E-05	***6.28E+00***	***6.27E+00***	2.28E+07
83Br	8.64E+03	8.02E-05	***5.40E-01***	***5.76E-01***	1.99E+06
84Br	1.91E+03	3.63E-04	***9.67E-01***	***1.01E+00***	3.54E+06
84mBr	3.60E+02	1.93E-03	***2.27E-02***	***1.78E-02***	7.83E+04
85Br	1.72E+02	4.03E-03	***1.30E+00***	***1.36E+00***	4.75E+06
86Br	5.55E+01	1.25E-02	***1.82E+00***	***1.73E+00***	6.52E+06
87Br	5.59E+01	1.24E-02	***2.07E+00***	***2.11E+00***	7.54E+06
88Br	1.64E+01	4.23E-02	***1.74E+00***	***2.12E+00***	6.61E+06
89Br	4.40E+00	1.58E-01	***1.08E+00***	***1.44E+00***	4.21E+06
90Br	1.90E+00	3.65E-01	***5.64E-01***	***7.47E-01***	2.19E+06
91Br	5.40E-01	1.28E+00	***2.24E-01***	***1.37E-01***	7.43E+05
92Br	3.40E-01	2.04E+00	***2.68E-02***	***2.09E-02***	9.25E+04
93Br	1.00E-01	6.93E+00	***3.08E-03***	***6.41E-02***	5.95E+04
96Br	2.00E-02	3.47E+01	***1.91E-06***	***2.28E-07***	5.59E+00

Maximum saturation activities calculated for a mass of 1 g of any fissionable material using the maximum cumulative fission yields and fission-cross sections irradiated by a thermal fluence rate of 7 E12 $cm^{-2}s^{-1}$ and a non-thermal fluence rate of 2 E12 $cm^{-2}s^{-1}$ are as follows:

Nuclide	Half-Life s	Decay Constant 1/s	Cumulative Yield % Thermal Fission	Cumulative Yield % Non- Thermal Fission	Calculated Activity, uCi
83mKr	6.70E+03	1.04E-04	***5.59E+00***	***2.22E+00***	1.50E+08
85mKr	1.61E+04	4.30E-05	***5.63E+00***	***3.47E+00***	1.60E+08
85Kr	3.39E+08	2.05E-09	***5.71E+00***	***3.47E+00***	1.62E+08
87Kr	4.57E+03	1.52E-04	***7.18E+00***	***6.95E+00***	2.21E+08
88Kr	1.02E+04	6.78E-05	***7.81E+00***	***6.88E+00***	2.36E+08
89Kr	1.89E+02	3.67E-03	***8.66E+00***	***7.47E+00***	2.61E+08
90Kr	3.23E+01	2.15E-02	***6.37E+00***	***7.96E+00***	2.09E+08
91Kr	8.60E+00	8.06E-02	***3.54E+00***	***6.25E+00***	1.29E+08
92Kr	1.84E+00	3.77E-01	***1.69E+00***	***4.12E+00***	6.91E+07
93Kr	1.29E+00	5.37E-01	***5.23E-01***	***1.87E+00***	2.55E+07
94Kr	2.10E-01	3.30E+00	***1.95E-01***	***7.34E-01***	9.76E+06
95Kr	7.80E-01	8.89E-01	***3.71E-02***	***1.27E-01***	1.77E+06
96Kr	8.00E-02	8.66E+00	***3.78E-02***	***3.42E-02***	1.15E+06
97Kr	6.30E-02	1.10E+01	***1.61E-03***	***9.31E-04***	4.53E+04
131mXe	1.03E+06	6.74E-07	***4.21E-02***	***4.54E-02***	1.33E+06
133mXe	1.89E+05	3.66E-06	***1.99E-01***	***1.97E-01***	6.16E+06
133Xe	4.53E+05	1.53E-06	***7.02E+00***	***6.97E+00***	2.17E+08
135mXe	9.18E+02	7.55E-04	***1.22E+00***	***1.15E+00***	3.75E+07
135Xe	3.28E+04	2.12E-05	***7.70E+00***	***7.54E+00***	2.38E+08
137Xe	2.29E+02	3.02E-03	***6.84E+00***	***6.58E+00***	2.10E+08
138Xe	8.46E+02	8.19E-04	***7.19E+00***	***6.27E+00***	2.17E+08
139Xe	3.97E+01	1.75E-02	***6.93E+00***	***6.90E+00***	2.15E+08
140Xe	1.36E+01	5.10E-02	***4.22E+00***	***6.77E+00***	1.48E+08
141Xe	1.72E+00	4.03E-01	***2.11E+00***	***4.41E+00***	8.13E+07
142Xe	1.22E+00	5.68E-01	***8.27E-01***	***2.29E+00***	3.58E+07
143Xe	3.00E-01	2.31E+00	***1.94E-01***	***7.03E-01***	9.53E+06
144Xe	1.20E+00	5.78E-01	***3.14E-02***	***2.34E-01***	2.37E+06
145Xe	1.88E-01	3.69E+00	***2.57E-03***	***1.98E-02***	1.98E+05
131I	6.93E+05	1.00E-06	***3.86E+00***	***4.17E+00***	1.22E+08
132I	8.26E+03	8.39E-05	***5.41E+00***	***5.33E+00***	1.67E+08
133I	7.49E+04	9.26E-06	***7.02E+00***	***6.97E+00***	2.17E+08
134I	3.16E+03	2.20E-04	***7.82E+00***	***7.98E+00***	2.44E+08
135I	2.37E+04	2.93E-05	***7.42E+00***	***6.99E+00***	2.27E+08
83Br	8.64E+03	8.02E-05	***5.59E+00***	***2.22E+00***	1.50E+08
84Br	1.91E+03	3.63E-04	***9.20E+00***	***4.09E+00***	2.50E+08
84mBr	3.60E+02	1.93E-03	***3.13E-01***	***4.19E-01***	1.04E+07
85Br	1.72E+02	4.03E-03	***5.61E+00***	***3.47E+00***	1.59E+08
86Br	5.55E+01	1.25E-02	***6.01E+00***	***5.89E+00***	1.86E+08
87Br	5.59E+01	1.24E-02	***4.78E+00***	***6.80E+00***	1.62E+08
88Br	1.64E+01	4.23E-02	***2.72E+00***	***6.03E+00***	1.07E+08
89Br	4.40E+00	1.58E-01	***1.31E+00***	***5.02E+00***	6.62E+07
90Br	1.90E+00	3.65E-01	***5.53E-01***	***2.62E+00***	3.14E+07
91Br	5.40E-01	1.28E+00	***2.25E-01***	***1.01E+00***	1.24E+07
92Br	3.40E-01	2.04E+00	***3.83E-02***	***2.34E-01***	2.54E+06
93Br	1.00E-01	6.93E+00	***8.31E-03***	***6.42E-02***	6.43E+05
96Br	2.00E-02	3.47E+01	***1.91E-06***	***3.52E-05***	2.89E+02

RELEASED ACTIVITY

Source dispersal fractions, D, are applied to estimate released activity into the reactor bay. The dispersed activity, $A_D(t)$, is given by the following:

$$\mathbf{A_D(\infty) = A(\infty)D} \qquad \boxed{\text{EQ 2}}$$

Activity (A) is in decays per second, dps, i.e. Bq and may be converted to other activity units, e.g. uCi using the conversion factor of 3.7E4 dps = 1 uCi

Dispersal Fraction, D

Precautions are taken to limit the release and potential intake of radioactive materials from inhalation. Normal precautions for fueled experiments would include encapsulation and shielding.

Failure of the encapsulation by non-violent means is the credible scenario considered in this analysis. Due to the production of gases and vapors, fueled experiments are to be doubly encapsulated as required by the facility TS. Failure of two layers of encapsulation would be needed for a release to occur and is considered unlikely. Additionally, intervening shielding would offer some reduction in the release of particulates. It is also normal practice to allow irradiated samples to decay prior to being handled and for personnel not to be continuously present at the experiment location during irradiation.

Furthermore, samples are limited to non-volatile materials with particle sizes greater than 10 microns thereby resulting in negligible release of respirable particulates [Ref 4]. Fissionable material in the form of a gas or volatile or combustible material is not allowed.

As a result, only the release of radioactive noble gases and halogens into the minimum free volume of the reactor building is considered from failure of the two layers of encapsulation.

From the above discussion, values for the dispersal fraction (D) are taken from References 1, 2, and 3. D values used are as follows:

Form	Dispersal Fraction, D
Gas	1
Halogen	1
Particulate	0

CONCENTRATION and TIME INTEGRATED EXPOSURE

The sample is assumed to contain saturated activities of radioactive fission gases (Kr, Xe) and halogens (I, Br) at the time of encapsulation failure. All of the fission gas and halogen radioactivity is assumed to be instantaneously released and uniformly mixed into the minimum free reactor bay volume resulting in uniform airborne activity distribution throughout the entire reactor bay.

The initial released concentration, C(0), in the reactor bay is given by the following:

$$\mathbf{C(0) = \frac{A_D(\infty)}{V}} \qquad \text{EQ3}$$

where, V of 2.25E9 ml is the minimum reactor bay free air volume in the experimental facility area reported in Reference 12.

Time Integrated Exposures for Fission Gas and Halogens Inside the Reactor Building

The time-integrated exposure (or time averaged concentration) with removal by radioactive decay and ventilation system inside the reactor building was calculated as follows:

$$\mathbf{\Omega = \int C(0)\, e^{-kt} dt = C(0)\, [\, 1- e^{-kT}] \,/\, k} \qquad \text{EQ 4}$$

where, Ω is the time integrated exposure in uCi-h/ml
$k = \lambda + v$ in h^{-1}
v is the confinement ventilation mode air removal rate constant in h^{-1}
v = 1.26 E-4 s^{-1} or 0.453 h^{-1} at a 600 cfm exhaust rate in confinement
t is exposure time, with limits of integration from 0 to T, in hours
T is the evacuation time of 0.05 hours (3 minutes) inside the reactor building

Time Integrated Exposures for Particulates Inside the Reactor Building:

Particulate activity of Rb and Cs are produced after the release followed by decay of Kr and Xe radionuclides, respectively. While inside the reactor building, particulate activity may buildup.

Upon review it is noted that the parent nuclide (1) have shorter half-lives than the decay product (2), thereby resulting in a non-equilibrium condition.

Solving for the time present in the reactor building at time “t” after release gives the following [Ref 20,21,22] for the activity of the decay product:

$$\mathbf{A_2(t) = A_1(0)\, k_2 \,/\, (k_1 - k_2)\, [e^{-k2t} - e^{-k1t}]} \qquad \text{EQ 5}$$

Similar to EQ 4, the time-integrated exposure with removal by radioactive decay and ventilation system inside the reactor building was calculated as follows for the decay products:

$$\Omega_2 = A_1(0)[k_2 / (k_1 - k_2)]\ [(1-e^{-k2T})/k_2\ -\ (1-e^{-k1T})/k_1] \qquad \text{EQ 6}$$

where T is the evacuation time of 0.05 hours (3 minutes)

Time Integrated Exposures for Fission Gas and Halogens Outside the Reactor Building

Time-integrated exposure in public areas is further reduced by removal of halogens and particulates by the confinement filters and by atmospheric dispersion.

Time-integrated exposure outside the reactor building was calculated as follows:

$$\Omega_p = \int C(0)\ e^{-kt}(1-R)[X/Q]\ F\ dt \qquad \text{EQ 7}$$

$$\Omega_p = C(0)\ (1-R)[X/Q]\ F[\ 1- e^{-kT}]\ /\ k \qquad \text{EQ 8}$$

where, Ω_p is the time integrated exposure in uCi-h/ml for members of the public
C(0) is in uCi/ml
k is in h^{-1}
t is exposure time in hours
R = 0.9 for halogens and R = 0 for noble gases
F is the volumetric stack exhaust rate = 0.283 m^3/s in confinement ventilation mode, converted from 600 cfm
[X/Q] is the most limiting atmospheric dispersion parameter for 2 hours or 24 hours
T is 2 hours or 24 hours for public exposure time outside the reactor building

For areas outside the reactor building, Rb and Cs particulate activity is negligible due to decay over the release time of 2 hours or 24 hours (public exposure time), filtration, and atmospheric dispersion.

Filter Retention [Ref 5]

Filter retention (R) for the ventilation system filters and radionuclides are summarized below:

Exhaust Mode and Filter	Noble Gas Retention	Particulate Retention (R) for > 0.3 microns	Iodine Retention (R)
Confinement HEPA	0	0.9997	0
Confinement Charcoal	0	0	0.9

High Efficiency Particulate Absorbers (HEPA) and charcoal beds are used in the confinement mode of ventilation. Filter removal is given by the product (1-R), where R is the retention factor.

Testing is performed per TS 4.5 on the ventilation system, including filter testing in accordance with TS 4.5 e. Maintenance and surveillance is in place for testing of the ventilation system. Testing methods follow ASME N510-1989 "Testing of Nuclear Air Treatment Systems". Testing is also required following major maintenance of the filters or housing. Testing and maintenance are documented in facility surveillance files as required by TS 6.4 and 6.8.

- Acceptance criteria are 99.97 percent for HEPA tested with 0.3 micron aerosols and 99 percent for charcoal tested with freon R-11.
- Charcoal filters are tested by the vendor prior to installation in the confinement system and have a reported retention factor of 0.99 for methyl iodine. A filter retention factor of 0.9 is used for iodine for conservatism.

ATMOSPHERIC DISPERSION [Ref 1,13, 23]

Atmospheric dispersion is assessed using methods given in References 1, 13 and 23 from 100 m to 5000 m away from the reactor stack.

Atmospheric dispersion is defined by parameter [X/Q]. [X/Q] is the ratio of the airborne activity concentration at a given location to the activity exhaust rate. [X/Q] equations presented below are considered applicable to locations that are at or beyond 100 m from the reactor stack.

The general equation for [X/Q] is as follows:

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z u}[e^{-\frac{y^2}{2\sigma_y^2}}][e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \boxed{\text{EQ 9}}$$

where, $[X/Q]_{x.y.z}$ is the atmospheric dispersion parameter for location (x,y,z) in s/m^3
X is in Ci/m^3 and Q is in Ci/s
x is the downwind distance from the stack to receptor in m
y is the lateral distance from the plume centerline in m
z is the receptor elevation in m
σ_y is the lateral dispersion parameter in m for PG weather stability classes
σ_z is the vertical dispersion parameter in m for PG weather stability classes
h is the physical stack height in m, or 30 m
μ is wind speed in m/s

NOTES: z and h are relative to the ground elevation of 0 m

Radioactive decay during transport is neglected.

The real (z-h) and a totally reflected plume (z+h) from the ground surface are included in the [X/Q] general equation

Dispersion Parameters

Pasquill-Gifford (PG) weather stability classes A through F are used for [X/Q] and are characterized by the following:

- σ_y is the lateral dispersion parameter in m
- σ_z is the vertical dispersion parameter in m

Dispersion parameters σ_y and σ_z were calculated using fitting data from NUREG 1887 "RASCAL 3.0.5: Description of Models and Methods" [Ref 24]. These calculated dispersion parameters for weather stability classes A through F were used to in the [X/Q] equations.

Release Rate

Release rate, Q, is calculated from the measured concentration in the reactor exhaust duct at a given stack exhaust rate, either in confinement ventilation mode of operation:

$$\mathbf{Q = C\ F} \qquad \boxed{\text{EQ 10}}$$

where, Q is the release rate in Ci/s
C is the concentration in Ci/m^3 and F is the stack exhaust in m^3/s
F = 0.283 m^3/s in confinement

Stack Height

EQ 9 given above may be modified for effective stack height. Stack height is affected by momentum effects due to the velocity of the exhausted air, buoyant effects due to the temperature of the exhausted air, and the exhaust velocity relative to the wind speed.

ANSI/ANS-15.7 [Ref 1] guidance for the effective stack height is not applicable since the exhaust velocity is less than 10 m/s in all ventilation modes and the temperature difference of the air exhaust and ambient air is well below 50 degrees C.

US NRC Regulatory Guide 1.111 [Ref 31] states that the effective stack height for effluents exhausted from release points more than twice the height of surrounding solid structures is determined as follows:

$$h_e = h + h_{pr} - h_t - c$$

where,

h_e is the effective stack height
h is the physical stack height
h_{pr} is the plume rise above the release point due to buoyancy effects and momentum
h_t is the difference in terrain height between the release point and the location of interest which must be greater than 0 meters.
c is the downwash correction factor

It is noted that for the reactor facility:

- h_e occurs at distances away from the stack release point and is assumed to be reached at a distance equal to 10 times the stack height, or 300 m. For distances less than 300 m the change in the physical stack height, Δh, is taken at 0 m giving h_e = h, or 30 m.
- h is 30 m (100 feet) and it is noted that the stack slightly exceeds 2 times the height of surrounding structures and buildings
- h_t is taken as 0 m, since there are no major valleys or hills nearby
- h_{pr} is based on the following:
  - momentum effects due to the exhaust velocity
  - heat emission rate is taken as 0 since the temperature difference of the air exhaust and ambient air is well below 50 degrees C
  - the equations given in the references from Regulatory Guide 1.111 or ANSI/ANS 15.7
- c is applicable if the exit velocity, V, is less than 1.5 times the horizontal wind speed, u. c is 0 m if V is equal to or exceeds 1.5u, i.e. if c is not positive.

Following guidance and equations given in US NRC Regulatory Guide 1.111, the downwash factor is applicable in confinement mode. V/u has a value of 1.44 giving a value of 0.09 m for c. The downwash correction does not significantly affect effective stack height.

From ANSI/ANS 15.7, the effective stack height is given by the following for exhaust with temperatures less than 50 degrees C:

$$\mathbf{h_e = h + d[V/u]^{1.4}} \qquad \boxed{\text{EQ 11}}$$

where, d is the stack diameter in m (0.5 m)
V is the exhaust velocity in m/s = 1.41 m/s in confinement mode
μ is the horizontal wind speed in m/s = 1 m/s by assumption

Effective stack height for release at a wind speed of 1 m/s were calculated to be slightly greater than the physical stack height (31 to 33 m vs 30 m) for confinement mode.

Stack heights greater than 30 m give lower [X/Q] values. For simplification and conservatism, the actual stack height of 30 m is used to calculate [X/Q], i.e. effective stack height is not used for [X/Q] calculations.

Release Time

Release time is taken as either 2 hours or 24 hours for dose assessment to members of the public. References 1 and 3 indicate times of 2 hours or more are to be used for public dose assessment.

Release Time of 2 Hours or Less

For a release of 2 hours or less it is assumed that the weather stability class, wind speed, and wind direction remain constant. Assumptions made are as follows:

- The assumed wind speed (μ) from ANSI/ANS-15.7[Ref 1] is 1 m/s
- The most restrictive weather stability class for the given location is used
- The receptor is assumed to be on the plume centerline, i.e. y = 0 m

With the noted assumptions, equation EQ 9 becomes:

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z}\left[e^{\left[-\frac{(z-h)^2}{2\sigma_z^2}\right]} + e^{\left[-\frac{(z+h)^2}{2\sigma_z^2}\right]}\right] \qquad \text{EQ 12}$$

The plume centerline equation above accounts for a receptor location at any elevation relative to the ground level. If the receptor is at ground level, i.e. z = 0 m, then equation EQ 12 becomes:

$$[X/Q]_{x,y,z} = \frac{1}{\pi\sigma_y\sigma_z}\left[e^{\left(-\frac{h^2}{2\sigma_z^2}\right)}\right] \qquad \text{EQ 13}$$

where, the real and totally reflected plume are summed

Release Time of 2 Hours or Longer

EQ 9 given above may be modified for sector averaging. Sector averaging applies if the wind direction deviates sufficiently across the sector with increased time. Averaging over a sector width, i.e.the lateral direction, or “y” dimension, is used to represent a meandering plume.

Sector averaging is considered valid at downwind distances (x) if $\pi x/n > 2\sigma_y$ [Ref 13] and for periods greater than 2 hours [Ref 1].

On inspection for the reactor facility stack height where the relationship $\pi x/n > 2\sigma_y$ is valid for n = 16 gives the following minimum distances for sector averaging for PG weather stability classes A through F:

x (m)	π x / n	2 σy A	2 σy B	2 σy C	2 σy D	2 σy E	2 σy F
100	**1.96E+01**	4.68E+01	3.29E+01	2.67E+01	**1.88E+01**	**1.34E+01**	**9.24E+00**
500	9.81E+01	2.00E+02	1.41E+02	1.14E+02	**8.06E+01**	**5.73E+01**	**3.95E+01**
1000	1.96E+02	3.75E+02	2.63E+02	2.14E+02	**1.51E+02**	**1.07E+02**	**7.39E+01**
1500	2.94E+02	5.40E+02	3.80E+02	3.09E+02	**2.17E+02**	**1.54E+02**	**1.07E+02**
2000	3.93E+02	7.01E+02	4.92E+02	4.00E+02	**2.82E+02**	**2.00E+02**	**1.38E+02**
2500	**4.91E+02**	8.57E+02	6.02E+02	**4.89E+02**	**3.45E+02**	**2.45E+02**	**1.69E+02**
3000	5.89E+02	1.01E+03	7.10E+02	**5.77E+02**	**4.06E+02**	**2.89E+02**	**1.99E+02**
3500	6.87E+02	1.16E+03	8.16E+02	**6.63E+02**	**4.67E+02**	**3.32E+02**	**2.29E+02**
4000	7.85E+02	1.31E+03	9.21E+02	**7.48E+02**	**5.27E+02**	**3.75E+02**	**2.59E+02**
4500	8.83E+02	1.46E+03	1.02E+03	**8.32E+02**	**5.86E+02**	**4.17E+02**	**2.88E+02**
5000	9.81E+02	1.60E+03	1.13E+03	**9.15E+02**	**6.44E+02**	**4.58E+02**	**3.16E+02**
10000	1.96E+03	3.00E+03	2.11E+03	**1.71E+03**	**1.21E+03**	**8.57E+02**	**5.92E+02**
25000	**4.91E+03**	6.86E+03	**4.82E+03**	**3.92E+03**	**2.76E+03**	**1.96E+03**	**1.35E+03**
50000	9.81E+03	1.28E+04	**9.01E+03**	**7.32E+03**	**5.16E+03**	**3.67E+03**	**2.53E+03**

Stability Class	Minimum Distance (m)
A	>50,000
B	25,000
C	2500
D, E, F	100

The sector average model is as follows for any receptor elevation:

$$\overline{[X/Q]_{x,y,z}} = \sqrt{2/\pi}\,\frac{n}{2\pi x}\,\frac{f}{2\sigma_z u}\left[e^{\left[-\frac{(z-h)^2}{2\sigma_z^2}\right]} + e^{\left[-\frac{(z+h)^2}{2\sigma_z^2}\right]}\right] \qquad \text{EQ 14}$$

where, the sector average [X/Q] is $\overline{[X/Q]}$

f is the frequency fraction for wind direction and wind speed

Release Time From 2 to 24 Hours

The PG weather stability class frequency, wind direction frequency (f), and wind speed (μ) remain constant. The most restrictive PG weather stability class was used for a given downwind location (x,y,z). From ANSI/ANS-15.7 [Ref 1], f is set at 1 and μ is 1 m/s.

If sector averaging is not valid, EQ 15 (i.e. the same as EQ 12) was used for all elevations (z):

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \boxed{\text{EQ 15}}$$

If sector averaging is valid, EQ 14 was used for all elevations (z). Re-writing EQ 14 with the noted assumptions for f and μ gives the following:

$$\overline{[X/Q]_{x,y,z}} = \frac{2.032}{2\sigma_z x}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \boxed{\text{EQ 16}}$$

where, for 16 sectors (i.e. n = 16) then $(16 / 2\pi) [2 / \pi]^{1/2} = 2.032$

The following simplifications are made regarding releases 2 to 24 hours:

- Stability classes A, B, and C were not sector averaged at any distance greater than 100 m for conservatism. EQ 15 was used.
- Stability classes D, E, and F were sector averaged at distances greater than 100 m. EQ 16 was used.

[X/Q] was calculated using EQ 12, EQ 15, and EQ 16 using the following weather parameters for 2 hour and 24 hour release times:

Duration	PG Stability Class	PG Stability Frequency	Wind Frequency, f	Wind (m/s)	Lateral Direction (y in m)
< 2 h	A through F	1	1	1	0, centerline
2 h to 24 h	A, B, C	1	1	1	0, centerline
2 h to 24 h	D, E, F	1	1	1	Sector Averaged

Maximum [X/Q] Values

Using the above equations, the following maximum [X/Q] values were calculated for distances from 100 to 5000 m away from the reactor stack:

Time in Hours	Height (m)	Maximum X/Q (s/m^3)	Class / Distance (m)
< 2	0	1.7E-4	B / 200
	10	2.0E-4	B / 150
	20	6.5E-4	C / 100
	30	1.5E-2	F / 100
	40	6.5E-4	C / 100
	50	1.9E-4	B / 150
	60	8.5E-5	B / 200
2 to 24	0	1.7E-4	B / 250
	10	2.0E-4	B / 150
	20	6.5E-4	C / 100
	30	4.5E-3	F / 100
	40	6.5E-4	C / 100
	50	1.9E-4	B / 150
	60	8.5E-5	B / 200

The most restrictive PG weather stability class was used for a given downwind location (x,y,z) to calculate [X/Q].

The maximum [X/Q] values were 1.5E-2 s/m^3 and 4.5E-3 s/m^3 for release times of 2 hours and 24 hours, respectively.

Maximum [X/Q] values for 2 hours and 24 hours were then used to calculate time integrated exposures in EQ8. The resulting Ω_p values are used in EQ 17 to assess dose to members of the public.

TIME INTERGRATED EXPOSURE RESULTS

Using the above equations, the time integrated exposures from irradiation for the reference mass of 1 g of U-235 were calculated.

For public exposure time of 24 hours, the following data was used:

DATA					
Nuclide	***U235***		Target atoms, N	2.56E+21	atoms
Mass =	***1***	**g**	Thermal fission rate	1.05E+13	f/s
Mass Number, A =	***235***	**g/mol**	Non-thermal fission rate	2.93E+12	f/s
Sigma thermal	***585***	**b**	Total fission rate	1.34E+13	f/s
Sigma non-thermal	***571***	**b**	Reactor volume =	2.25E+09	ml
X/Q =	***4.53E-03***	**s/m3**	F confinement =	2.83E-01	m3/s
Public exposure =	***24***	**h**	v confinement =	1.26E-04	1/s
NOTE: Fluxes listed are maximum observed at 1 MW					
Thermal flux =	7.00E+12	cm-2s-1			
Non-thermal flux =	2.00E+12	cm-2s-1	Evacuation time =	0.05	h
			NG reactor correction =	1.00E-01	
			(1-R) halogens =	1.00E-01	
			(1-R) noble gas =	0	

For public exposure time of 2 hours, the following data was used:

DATA					
Nuclide	***U235***		Target atoms, N	2.56E+21	atoms
Mass =	***1***	**g**	Thermal fission rate	1.05E+13	f/s
Mass Number, A =	***235***	**g/mol**	Non-thermal fission rate	2.93E+12	f/s
Sigma thermal	***585***	**b**	Total fission rate	1.34E+13	f/s
Sigma non-thermal	***571***	**b**	Reactor volume =	2.25E+09	ml
X/Q =	***1.53E-02***	**s/m3**	F confinement =	2.83E-01	m3/s
Public exposure =	***2***	**h**	v confinement =	1.26E-04	1/s
NOTE: Fluxes listed are maximum observed at 1 MW					
Thermal flux =	7.00E+12	cm-2s-1			
Non-thermal flux =	2.00E+12	cm-2s-1	Evacuation time =	0.05	h
			NG reactor correction =	1.00E-01	
			(1-R) halogens =	1.00E-01	
			(1-R) noble gas =	0	

U-235 Time Integrated Exposures for Public Exposure Time of 24 hours:

Nuclide	Decay Constant 1/s	Calculated Activity, uCi	Intial Concentration C(0), uCi/ml	Time Integrated Exposures Reactor uCi-h/ml	Public uCi-h/ml
83mKr	1.04E-04	1.97E+06	8.78E-04	4.30E-05	1.36E-06
85mKr	4.30E-05	4.73E+06	2.10E-03	1.04E-04	4.44E-06
85Kr	2.05E-09	1.04E+06	4.61E-04	2.28E-05	1.30E-06
87Kr	1.52E-04	9.27E+06	4.12E-03	2.01E-04	5.29E-06
88Kr	6.78E-05	1.28E+07	5.68E-03	2.79E-04	1.05E-05
89Kr	3.67E-03	1.59E+07	7.08E-03	2.57E-04	6.65E-07
90Kr	2.15E-02	1.74E+07	7.74E-03	9.76E-05	1.28E-07
91Kr	8.06E-02	1.19E+07	5.29E-03	1.82E-05	2.33E-08
92Kr	3.77E-01	5.83E+06	2.59E-03	1.91E-06	2.45E-09
93Kr	5.37E-01	1.66E+06	7.38E-04	3.82E-07	4.90E-10
94Kr	3.30E+00	2.96E+05	1.31E-04	1.11E-08	1.42E-11
95Kr	8.89E-01	3.83E+04	1.70E-05	5.32E-09	6.82E-12
96Kr	8.66E+00	1.09E+05	4.85E-05	1.55E-09	1.99E-12
97Kr	1.10E+01	1.75E+02	7.79E-08	1.97E-12	2.52E-15
131mXe	6.74E-07	1.43E+05	6.35E-05	3.14E-06	1.79E-07
133mXe	3.66E-06	6.92E+05	3.08E-04	1.52E-05	8.46E-07
133Xe	1.53E-06	2.43E+07	1.08E-02	5.34E-04	3.02E-05
135mXe	7.55E-04	4.12E+06	1.83E-03	8.47E-05	7.41E-07
135Xe	2.12E-05	2.38E+07	1.06E-02	5.21E-04	2.56E-05
137Xe	3.02E-03	2.21E+07	9.83E-03	3.75E-04	1.11E-06
138Xe	8.19E-04	2.26E+07	1.00E-02	4.62E-04	3.79E-06
139Xe	1.75E-02	1.76E+07	7.84E-03	1.19E-04	1.59E-07
140Xe	5.10E-02	1.25E+07	5.55E-03	3.01E-05	3.87E-08
141Xe	4.03E-01	4.25E+06	1.89E-03	1.30E-06	1.67E-09
142Xe	5.68E-01	1.69E+06	7.53E-04	3.68E-07	4.72E-10
143Xe	2.31E+00	1.64E+05	7.27E-05	8.74E-09	1.12E-11
144Xe	5.78E-01	3.42E+04	1.52E-05	7.30E-09	9.36E-12
145Xe	3.69E+00	8.75E+02	3.89E-07	2.93E-11	3.76E-14
131I	1.00E-06	1.07E+07	4.77E-03	2.36E-04	1.34E-06
132I	8.39E-05	1.59E+07	7.07E-03	3.47E-04	1.20E-06
133I	9.26E-06	2.43E+07	1.08E-02	5.34E-04	2.85E-06
134I	2.20E-04	2.82E+07	1.26E-02	6.08E-04	1.29E-06
135I	2.93E-05	2.28E+07	1.01E-02	4.99E-04	2.32E-06
83Br	8.02E-05	1.99E+06	8.82E-04	4.33E-05	1.53E-07
84Br	3.63E-04	3.54E+06	1.57E-03	7.53E-05	1.15E-07
84mBr	1.93E-03	7.83E+04	3.48E-05	1.46E-06	6.05E-10
85Br	4.03E-03	4.75E+06	2.11E-03	7.44E-05	1.81E-08
86Br	1.25E-02	6.52E+06	2.90E-03	5.72E-05	8.18E-09
87Br	1.24E-02	7.54E+06	3.35E-03	6.65E-05	9.53E-09
88Br	4.23E-02	6.61E+06	2.94E-03	1.92E-05	2.47E-09
89Br	1.58E-01	4.21E+06	1.87E-03	3.30E-06	4.23E-10
90Br	3.65E-01	2.19E+06	9.73E-04	7.41E-07	9.50E-11
91Br	1.28E+00	7.43E+05	3.30E-04	7.15E-08	9.17E-12
92Br	2.04E+00	9.25E+04	4.11E-05	5.60E-09	7.18E-13
93Br	6.93E+00	5.95E+04	2.64E-05	1.06E-09	1.36E-13
96Br	3.47E+01	5.59E+00	2.48E-09	1.99E-14	2.55E-18
Rb88	6.53E-04			9.23E-07	
Rb89	7.50E-04			7.72E-07	
Rb90	4.44E-03			5.63E-07	
Rb91	1.28E+00			6.26E-08	
Rb92	1.55E-01			1.41E-09	
Rb93	1.18E-01			1.97E-10	
Rb94	2.56E-01			9.30E-13	
Rb95	1.84E+00			1.66E-12	
Rb96	3.48E+00			4.98E-14	
Cs138	3.46E-04			9.01E-07	
Cs139	1.24E-03			2.92E-07	
Cs140	1.09E-02			1.35E-07	
Cs141	2.78E-02			8.91E-10	
Cs142	4.12E-01			1.80E-10	
Cs143	3.89E-01			1.05E-12	
Cs144	6.86E-01			3.51E-12	
Cs145	1.17E+00			2.21E-15	

U-235 Time Integrated Exposures for Public Exposure Time of 2 hours:

Nuclide	Decay Constant 1/s	Calculated Activity, uCi	Intial Concentration C(0), uCi/ml	Time Integrated Exposures Reactor uCi-h/ml	Public uCi-h/ml
83mKr	1.04E-04	1.97E+06	8.78E-04	4.30E-05	3.72E-06
85mKr	4.30E-05	4.73E+06	2.10E-03	1.04E-04	1.05E-05
85Kr	2.05E-09	1.04E+06	4.61E-04	2.28E-05	2.63E-06
87Kr	1.52E-04	9.27E+06	4.12E-03	2.01E-04	1.54E-05
88Kr	6.78E-05	1.28E+07	5.68E-03	2.79E-04	2.66E-05
89Kr	3.67E-03	1.59E+07	7.08E-03	2.57E-04	2.25E-06
90Kr	2.15E-02	1.74E+07	7.74E-03	9.76E-05	4.32E-07
91Kr	8.06E-02	1.19E+07	5.29E-03	1.82E-05	7.88E-08
92Kr	3.77E-01	5.83E+06	2.59E-03	1.91E-06	8.28E-09
93Kr	5.37E-01	1.66E+06	7.38E-04	3.82E-07	1.65E-09
94Kr	3.30E+00	2.96E+05	1.31E-04	1.11E-08	4.79E-11
95Kr	8.89E-01	3.83E+04	1.70E-05	5.32E-09	2.30E-11
96Kr	8.66E+00	1.09E+05	4.85E-05	1.55E-09	6.73E-12
97Kr	1.10E+01	1.75E+02	7.79E-08	1.97E-12	8.52E-15
131mXe	6.74E-07	1.43E+05	6.35E-05	3.14E-06	3.61E-07
133mXe	3.66E-06	6.92E+05	3.08E-04	1.52E-05	1.73E-06
133Xe	1.53E-06	2.43E+07	1.08E-02	5.34E-04	6.13E-05
135mXe	7.55E-04	4.12E+06	1.83E-03	8.47E-05	2.50E-06
135Xe	2.12E-05	2.38E+07	1.06E-02	5.21E-04	5.64E-05
137Xe	3.02E-03	2.21E+07	9.83E-03	3.75E-04	3.76E-06
138Xe	8.19E-04	2.26E+07	1.00E-02	4.62E-04	1.28E-05
139Xe	1.75E-02	1.76E+07	7.84E-03	1.19E-04	5.36E-07
140Xe	5.10E-02	1.25E+07	5.55E-03	3.01E-05	1.31E-07
141Xe	4.03E-01	4.25E+06	1.89E-03	1.30E-06	5.64E-09
142Xe	5.68E-01	1.69E+06	7.53E-04	3.68E-07	1.59E-09
143Xe	2.31E+00	1.64E+05	7.27E-05	8.74E-09	3.79E-11
144Xe	5.78E-01	3.42E+04	1.52E-05	7.30E-09	3.16E-11
145Xe	3.69E+00	8.75E+02	3.89E-07	2.93E-11	1.27E-13
131I	1.00E-06	1.07E+07	4.77E-03	2.36E-04	2.71E-06
132I	8.39E-05	1.59E+07	7.07E-03	3.47E-04	3.16E-06
133I	9.26E-06	2.43E+07	1.08E-02	5.34E-04	5.98E-06
134I	2.20E-04	2.82E+07	1.26E-02	6.08E-04	4.01E-06
135I	2.93E-05	2.28E+07	1.01E-02	4.99E-04	5.28E-06
83Br	8.02E-05	1.99E+06	8.82E-04	4.33E-05	3.98E-07
84Br	3.63E-04	3.54E+06	1.57E-03	7.53E-05	3.76E-07
84mBr	1.93E-03	7.83E+04	3.48E-05	1.46E-06	2.04E-09
85Br	4.03E-03	4.75E+06	2.11E-03	7.44E-05	6.13E-08
86Br	1.25E-02	6.52E+06	2.90E-03	5.72E-05	2.76E-08
87Br	1.24E-02	7.54E+06	3.35E-03	6.65E-05	3.22E-08
88Br	4.23E-02	6.61E+06	2.94E-03	1.92E-05	8.34E-09
89Br	1.58E-01	4.21E+06	1.87E-03	3.30E-06	1.43E-09
90Br	3.65E-01	2.19E+06	9.73E-04	7.41E-07	3.21E-10
91Br	1.28E+00	7.43E+05	3.30E-04	7.15E-08	3.10E-11
92Br	2.04E+00	9.25E+04	4.11E-05	5.60E-09	2.43E-12
93Br	6.93E+00	5.95E+04	2.64E-05	1.06E-09	4.59E-13
96Br	3.47E+01	5.59E+00	2.48E-09	1.99E-14	8.62E-18
Rb88	6.53E-04			9.23E-07	
Rb89	7.50E-04			7.72E-07	
Rb90	4.44E-03			5.63E-07	
Rb91	1.28E+00			6.26E-08	
Rb92	1.55E-01			1.41E-09	
Rb93	1.18E-01			1.97E-10	
Rb94	2.56E-01			9.30E-13	
Rb95	1.84E+00			1.66E-12	
Rb96	3.48E+00			4.98E-14	
Cs138	3.46E-04			9.01E-07	
Cs139	1.24E-03			2.92E-07	
Cs140	1.09E-02			1.35E-07	
Cs141	2.78E-02			8.91E-10	
Cs142	4.12E-01			1.80E-10	
Cs143	3.89E-01			1.05E-12	
Cs144	6.86E-01			3.51E-12	
Cs145	1.17E+00			2.21E-15	

Using the above equations, the time integrated exposures from irradiation for the reference mass of 1 g of any fissionable material were calculated.

For public exposure time of 24 hours, the following data was used:

DATA					
Nuclide	***Any***		Target atoms, N	2.40E+21	atoms
Mass =	***1***	**g**	Thermal fission rate	8.93E+13	f/s
Mass Number, A =	***251***	**g/mol**	Non-thermal fission rate	2.55E+13	f/s
Sigma thermal	***5320***	**b**	Total fission rate	1.15E+14	f/s
Sigma non-thermal	***5310***	**b**	Reactor volume =	2.25E+09	ml
X/Q =	***4.53E-03***	**s/m3**	F confinement =	2.83E-01	m3/s
Public exposure =	***24***	**s**	v confinement =	1.26E-04	1/s
NOTE: Fluxes listed are maximum observed at 1 MW					
Thermal flux =	7.00E+12	cm-2s-1			
Non-thermal flux =	2.00E+12	cm-2s-1	Evacuation time =	0.05	h
			NG reactor correction =	1.00E-01	
			(1-R) halogens =	1.00E-01	
			(1-R) noble gas =	0	

For public exposure time of 2 hours, the following data was used:

DATA					
Nuclide	***Any***		Target atoms, N	2.40E+21	atoms
Mass =	***1***	**g**	Thermal fission rate	8.93E+13	f/s
Mass Number, A =	***251***	**g/mol**	Non-thermal fission rate	2.55E+13	f/s
Sigma thermal	***5320***	**b**	Total fission rate	1.15E+14	f/s
Sigma non-thermal	***5310***	**b**	Reactor volume =	2.25E+09	ml
X/Q =	***1.53E-02***	**s/m3**	F confinement =	2.83E-01	m3/s
Public exposure =	***2***	**h**	v confinement =	1.26E-04	1/s
NOTE: Fluxes listed are maximum observed at 1 MW					
Thermal flux =	7.00E+12	cm-2s-1			
Non-thermal flux =	2.00E+12	cm-2s-1	Evacuation time =	0.05	h
			NG reactor correction =	1.00E-01	
			(1-R) halogens =	1.00E-01	
			(1-R) noble gas =	0	

Time Integrated Exposures for Public Exposure Time of 24 hours for any fissionable material:

Nuclide	Decay Constant 1/s	Calculated Activity, uCi	Intial Concentration C(0), uCi/ml	Time Integrated Exposures Reactor uCi-h/ml	Public uCi-h/ml
83mKr	1.04E-04	1.50E+08	6.68E-02	3.27E-03	1.04E-04
85mKr	4.30E-05	1.60E+08	7.10E-02	3.50E-03	1.50E-04
85Kr	2.05E-09	1.62E+08	7.19E-02	3.55E-03	2.04E-04
87Kr	1.52E-04	2.21E+08	9.83E-02	4.79E-03	1.26E-04
88Kr	6.78E-05	2.36E+08	1.05E-01	5.16E-03	1.93E-04
89Kr	3.67E-03	2.61E+08	1.16E-01	4.20E-03	1.09E-05
90Kr	2.15E-02	2.09E+08	9.27E-02	1.17E-03	1.53E-06
91Kr	8.06E-02	1.29E+08	5.71E-02	1.97E-04	2.52E-07
92Kr	3.77E-01	6.91E+07	3.07E-02	2.26E-05	2.90E-08
93Kr	5.37E-01	2.55E+07	1.13E-02	5.87E-06	7.52E-09
94Kr	3.30E+00	9.76E+06	4.34E-03	3.65E-07	4.68E-10
95Kr	8.89E-01	1.77E+06	7.87E-04	2.46E-07	3.15E-10
96Kr	8.66E+00	1.15E+06	5.10E-04	1.64E-08	2.10E-11
97Kr	1.10E+01	4.53E+04	2.01E-05	5.08E-10	6.52E-13
131mXe	6.74E-07	1.33E+06	5.91E-04	2.92E-05	1.66E-06
133mXe	3.66E-06	6.16E+06	2.74E-03	1.35E-04	7.53E-06
133Xe	1.53E-06	2.17E+08	9.66E-02	4.78E-03	2.70E-04
135mXe	7.55E-04	3.75E+07	1.67E-02	7.71E-04	6.74E-06
135Xe	2.12E-05	2.38E+08	1.06E-01	5.21E-03	2.56E-04
137Xe	3.02E-03	2.10E+08	9.35E-02	3.57E-03	1.06E-05
138Xe	8.19E-04	2.17E+08	9.64E-02	4.43E-03	3.63E-05
139Xe	1.75E-02	2.15E+08	9.55E-02	1.45E-03	1.94E-06
140Xe	5.10E-02	1.48E+08	6.60E-02	3.59E-04	4.60E-07
141Xe	4.03E-01	8.13E+07	3.61E-02	2.49E-05	3.19E-08
142Xe	5.68E-01	3.58E+07	1.59E-02	7.77E-06	9.97E-09
143Xe	2.31E+00	9.53E+06	4.23E-03	5.09E-07	6.53E-10
144Xe	5.78E-01	2.37E+06	1.05E-03	5.06E-07	6.50E-10
145Xe	3.69E+00	1.98E+05	8.82E-05	6.64E-09	8.52E-12
131I	1.00E-06	1.22E+08	5.42E-02	2.68E-03	1.52E-05
132I	8.39E-05	1.67E+08	7.43E-02	3.65E-03	1.26E-05
133I	9.26E-06	2.17E+08	9.66E-02	4.77E-03	2.55E-05
134I	2.20E-04	2.44E+08	1.08E-01	5.25E-03	1.12E-05
135I	2.93E-05	2.27E+08	1.01E-01	4.98E-03	2.32E-05
83Br	8.02E-05	1.50E+08	6.68E-02	3.28E-03	1.15E-05
84Br	3.63E-04	2.50E+08	1.11E-01	5.32E-03	8.10E-06
84mBr	1.93E-03	1.04E+07	4.64E-03	1.94E-04	8.06E-08
85Br	4.03E-03	1.59E+08	7.08E-02	2.49E-03	6.08E-07
86Br	1.25E-02	1.86E+08	8.25E-02	1.63E-03	2.33E-07
87Br	1.24E-02	1.62E+08	7.21E-02	1.43E-03	2.05E-07
88Br	4.23E-02	1.07E+08	4.77E-02	3.12E-04	4.01E-08
89Br	1.58E-01	6.62E+07	2.94E-02	5.18E-05	6.65E-09
90Br	3.65E-01	3.14E+07	1.40E-02	1.06E-05	1.36E-09
91Br	1.28E+00	1.24E+07	5.50E-03	1.19E-06	1.53E-10
92Br	2.04E+00	2.54E+06	1.13E-03	1.54E-07	1.97E-11
93Br	6.93E+00	6.43E+05	2.86E-04	1.14E-08	1.47E-12
96Br	3.47E+01	2.89E+02	1.28E-07	1.03E-12	1.32E-16
Rb88	6.53E-04			1.70E-05	
Rb89	7.50E-04			1.26E-05	
Rb90	4.44E-03			6.74E-06	
Rb91	1.28E+00			6.76E-07	
Rb92	1.55E-01			1.67E-08	
Rb93	1.18E-01			3.03E-09	
Rb94	2.56E-01			3.07E-11	
Rb95	1.84E+00			7.69E-11	
Rb96	3.48E+00			5.25E-13	
Cs138	3.46E-04			8.64E-06	
Cs139	1.24E-03			3.56E-06	
Cs140	1.09E-02			1.61E-06	
Cs141	2.78E-02			1.70E-08	
Cs142	4.12E-01			3.80E-09	
Cs143	3.89E-01			6.12E-11	
Cs144	6.86E-01			2.43E-10	
Cs145	1.17E+00			5.01E-13	

Time Integrated Exposures for Public Exposure Time of 2 hours for any fissionable material:

Nuclide	Decay Constant 1/s	Calculated Activity, uCi	Intial Concentration C(0), uCi/ml	Time Integrated Exposures Reactor uCi-h/ml	Public uCi-h/ml
83mKr	1.04E-04	1.50E+08	6.68E-02	3.27E-03	2.83E-04
85mKr	4.30E-05	1.60E+08	7.10E-02	3.50E-03	3.56E-04
85Kr	2.05E-09	1.62E+08	7.19E-02	3.55E-03	4.10E-04
87Kr	1.52E-04	2.21E+08	9.83E-02	4.79E-03	3.69E-04
88Kr	6.78E-05	2.36E+08	1.05E-01	5.16E-03	4.90E-04
89Kr	3.67E-03	2.61E+08	1.16E-01	4.20E-03	3.67E-05
90Kr	2.15E-02	2.09E+08	9.27E-02	1.17E-03	5.17E-06
91Kr	8.06E-02	1.29E+08	5.71E-02	1.97E-04	8.51E-07
92Kr	3.77E-01	6.91E+07	3.07E-02	2.26E-05	9.81E-08
93Kr	5.37E-01	2.55E+07	1.13E-02	5.87E-06	2.54E-08
94Kr	3.30E+00	9.76E+06	4.34E-03	3.65E-07	1.58E-09
95Kr	8.89E-01	1.77E+06	7.87E-04	2.46E-07	1.07E-09
96Kr	8.66E+00	1.15E+06	5.10E-04	1.64E-08	7.09E-11
97Kr	1.10E+01	4.53E+04	2.01E-05	5.08E-10	2.20E-12
131mXe	6.74E-07	1.33E+06	5.91E-04	2.92E-05	3.36E-06
133mXe	3.66E-06	6.16E+06	2.74E-03	1.35E-04	1.54E-05
133Xe	1.53E-06	2.17E+08	9.66E-02	4.78E-03	5.48E-04
135mXe	7.55E-04	3.75E+07	1.67E-02	7.71E-04	2.27E-05
135Xe	2.12E-05	2.38E+08	1.06E-01	5.21E-03	5.65E-04
137Xe	3.02E-03	2.10E+08	9.35E-02	3.57E-03	3.57E-05
138Xe	8.19E-04	2.17E+08	9.64E-02	4.43E-03	1.23E-04
139Xe	1.75E-02	2.15E+08	9.55E-02	1.45E-03	6.54E-06
140Xe	5.10E-02	1.48E+08	6.60E-02	3.59E-04	1.55E-06
141Xe	4.03E-01	8.13E+07	3.61E-02	2.49E-05	1.08E-07
142Xe	5.68E-01	3.58E+07	1.59E-02	7.77E-06	3.37E-08
143Xe	2.31E+00	9.53E+06	4.23E-03	5.09E-07	2.21E-09
144Xe	5.78E-01	2.37E+06	1.05E-03	5.06E-07	2.19E-09
145Xe	3.69E+00	1.98E+05	8.82E-05	6.64E-09	2.88E-11
131I	1.00E-06	1.22E+08	5.42E-02	2.68E-03	3.08E-05
132I	8.39E-05	1.67E+08	7.43E-02	3.65E-03	3.32E-05
133I	9.26E-06	2.17E+08	9.66E-02	4.77E-03	5.35E-05
134I	2.20E-04	2.44E+08	1.08E-01	5.25E-03	3.46E-05
135I	2.93E-05	2.27E+08	1.01E-01	4.98E-03	5.27E-05
83Br	8.02E-05	1.50E+08	6.68E-02	3.28E-03	3.02E-05
84Br	3.63E-04	2.50E+08	1.11E-01	5.32E-03	2.66E-05
84mBr	1.93E-03	1.04E+07	4.64E-03	1.94E-04	2.72E-07
85Br	4.03E-03	1.59E+08	7.08E-02	2.49E-03	2.05E-06
86Br	1.25E-02	1.86E+08	8.25E-02	1.63E-03	7.87E-07
87Br	1.24E-02	1.62E+08	7.21E-02	1.43E-03	6.93E-07
88Br	4.23E-02	1.07E+08	4.77E-02	3.12E-04	1.35E-07
89Br	1.58E-01	6.62E+07	2.94E-02	5.18E-05	2.24E-08
90Br	3.65E-01	3.14E+07	1.40E-02	1.06E-05	4.60E-09
91Br	1.28E+00	1.24E+07	5.50E-03	1.19E-06	5.15E-10
92Br	2.04E+00	2.54E+06	1.13E-03	1.54E-07	6.65E-11
93Br	6.93E+00	6.43E+05	2.86E-04	1.14E-08	4.96E-12
96Br	3.47E+01	2.89E+02	1.28E-07	1.03E-12	4.45E-16
Rb88	6.53E-04			1.70E-05	
Rb89	7.50E-04			1.26E-05	
Rb90	4.44E-03			6.74E-06	
Rb91	1.28E+00			6.76E-07	
Rb92	1.55E-01			1.67E-08	
Rb93	1.18E-01			3.03E-09	
Rb94	2.56E-01			3.07E-11	
Rb95	1.84E+00			7.69E-11	
Rb96	3.48E+00			5.25E-13	
Cs138	3.46E-04			8.64E-06	
Cs139	1.24E-03			3.56E-06	
Cs140	1.09E-02			1.61E-06	
Cs141	2.78E-02			1.70E-08	
Cs142	4.12E-01			3.80E-09	
Cs143	3.89E-01			6.12E-11	
Cs144	6.86E-01			2.43E-10	
Cs145	1.17E+00			5.01E-13	

DOSE ASSESSMENT

EXTERNAL DOSE (other than submersion)

For radiological control purposes, external dose rates from are limited and controlled by facility procedures consistent with experimental limitations and conditions and 10 CFR Part 20 [Ref 25] requirements, including ALARA (As Low As Reasonably Achievable) practices.

All experiments require a radioactive materials authorization approved by the Reactor Safety and Audit Committee (RSAC) and the NCSU Radiation Safety Committee (RSC). Activity limits, handling conditions, and experimental uses by personnel are established in the radioactive material authorizations. [Ref 12]

Prior to conducting the irradiation, external dose rates and source activities are estimated, along with other required information, in the experiment request. Other information includes giving the materials and quantities present in the sample and indicating if any fissionable materials or materials with high cross-sections are present. The experiment request is reviewed and approved by reactor staff prior to conducting the irradiation. Fueled experiments are controlled as stated in TS 3.8 to include documented reviews, committee reviews by RSAC and RSC for new fueled experiments, and radiation monitoring.

The peak fission rate expected from fueled experiments is less than 5E12 f/s as compared to 3.1E16 f/s at 1 MW produced in the reactor fuel. For fueled experiments near the reactor core, this increases the measured radiation levels of approximately 1 mrem/h by less than 2 E-4 mrem/h (5E12 / 3.1E16) outside the reactor shield or at the top of the reactor pool.

Experimental beam tubes may be filled with various shielding materials (e.g. concrete, polyethylene, water, lead) to reduce external dose rates. Additional shielding may be placed outside the experimental beam tube, and within the reactor building, such as steel, concrete, and lead to reduce external dose rates. Appropriate access controls and monitoring as required by 10 CFR Part 20 [Ref25], the radiation protection program, and the radioactive/radiation authorization for the experiment are used to control personnel dose.

Radiation surveys are performed to verify that external dose rates, airborne activity, and contamination levels are within acceptable levels for new experiments, new experimental facilities or altered shielding. If needed, power ascension or an irradiation of short duration is performed to estimate dose rates at higher power and longer times.

Occupationally exposed personnel at the reactor facility are monitored as provided for in the facility radiation protection program. Doses to occupational personnel are limited administratively to doses below the regulatory limits as provided for in the facility radiation protection program.

Radiation monitoring with local alarms and Control Room notification, shielding, limitation of access/ stay times, boundary controls/distance (e.g. for High Radiation Area), alarming dosimeters, and use of approved procedures and Radiation Work Permits are included as needed to keep personnel dose ALARA.

All samples are surveyed at the time of removal from an experiment. Remote radiation monitoring may be used to measure and indicate the sample dose rate. Upon removal samples may be shielded or returned to storage for further decay as necessary. Samples removed are labeled and stored and used in areas which are posted as required by the facility radiation protection program.

As is the case with all experiments, and as directed by the facility radiation protection program, experiments producing abnormal, unexpected, or unacceptably high radiation levels are stopped. Radioactive material is shipped or disposed of as provided for in the facility radiation protection program.

Members of the public are not allowed into areas exceeding 2 mrem per hour or 100 mrem per year and also are not allowed to handle sources of radiation or radioactivity regardless of dose rate.

DOSE FROM RELEASED MATERIALS

Radiation dose from the submersion and inhalation pathways for the radioactive materials released include the following as defined in 10 CFR Part 20 [Ref25]:

- Deep dose-equivalent (DDE) from submersion
- Total effective dose-equivalent (TEDE) from inhalation and submersion given by the sum of the DDE from submersion and the committed effective dose-equivalent (CEDE) from inhalation
- Total organ dose-equivalent (TODE) given by the sum of the DDE from submersion and committed dose-equivalent (CDE) from inhalation for the organ of concern (thyroid)

Dose to occupational workers and members of the public is determined as follows for each radioactive material released:

$$\mathbf{D_{reference} = \Omega_D \cdot DCF} \qquad \text{EQ 17}$$

where, $D_{reference}$ is dose, in rem calculated for a reference mass of 1 gram at the reference fluence rates for a given radionuclide

Ω_D is the Time Integrated Exposure (uCi-h/ml).
$\Omega_D = \Omega$ from EQ 4 and Ω_2 from EQ 6 for the reactor building
$\Omega_D = \Omega_p$ from EQ 8 for areas outside the reactor building for public exposure times

DCF = Dose Conversion Factor in rem/h per uCi/ml converted from the concentrations listed in 10 CFR Part 20 Appendix B Table 1 and Table 2 [Ref 25] for submersion and inhalation doses. DCF has the units of rem/h per uCi/ml

$D_{reference}$ TEDE and TODE are summed for all nuclides to give the Total Reference Dose.

Dose Conversion Factors (DCF)

DCF are determined from 10 CFR Part 20 [Ref 25] Appendix B using the listed Derived Air Concentration (DAC), Annual Limit on Intake (ALI), or Effluent Concentration (EC) and directions given in Appendix B.

DCF were based on limiting values given in 10 CFR Part 20 [Ref 25] Appendix B, i.e. using the lower value listed for the different inhalation classes.

For inhalation, the listed DAC applies to the CDE limit of 50 rem per year if an organ is listed or to the CEDE limit of 5 rem per year if no organ is listed. When an organ is listed, the ALI listed in parentheses is used to determine the DAC associated with the CEDE as follows:

$$\textbf{DAC (effective dose) = (ALI in parentheses) / 2.4E9 ml} \qquad \text{EQ 18}$$

The listed EC applies to the effective dose limit of 0.1 rem per year for submersion DDE and 0.05 rem per year CEDE to an adult from inhalation. An age dependent factor of 2 is applied to the inhalation CEDE for other age groups.

For submersion DDE, an age dependent factor is not used.

DCF were calculated as follows:

For CEDE from inhalation:

$$\underset{\textit{for workers}}{\textit{Effective DCF}} = \frac{(5\ \text{rem} / 2000\ \text{h})}{[\text{10CFR20 Appendix B Table 1 DAC in uCi/ml}]} \qquad \text{EQ 19}$$

$$\underset{\textit{for public}}{\textit{Effective DCF}} = \frac{(0.05\ \text{rem} / 8760\ \text{h})(2)}{[\text{10CFR20 Appendix B Table 2 EC in uCi/ml}]} \qquad \text{EQ 20}$$

For DDE from submersion dose:

$$\underset{\textit{for workers}}{\textit{Effective DCF}} = \frac{(5\ \text{rem} / 2000\ \text{h})(0.1)}{[\text{10CFR20 Appendix B Table 1 DAC in uCi/ml}]} \qquad \text{EQ 21}$$

$$\underset{\textit{for public}}{\textit{Effective DCF}} = \frac{(0.1\ \text{rem} / 8760\ \text{h})}{[\text{10CFR20 Appendix B Table 2 EC in uCi/ml}]} \qquad \text{EQ 22}$$

For CDE for the thyroid from inhalation:

$$\underset{\textit{for workers}}{\textit{Organ DCF}} = \frac{(50\ \text{rem} / 2000\ \text{h})}{[\text{10CFR20 Appendix B Table 1 DAC in uCi/ml}]} \qquad \text{EQ 23}$$

Submersion Dose Correction

Reduction of submersion dose from photons emitted by radioactive noble gas fission products inside the reactor building is made based on room dimensions using the following [Ref 9,10,11, 13]:

$$\boldsymbol{f = f'Gk = u_{en}\ RGk} \qquad \boxed{\text{EQ 24}}$$

$$f = (4.92\text{E-}5/\text{cm})(905\ \text{cm})(2)(1.1) = 9.8\text{E-}2 \sim 0.1$$

Alternately, $f = 2k[1\text{-}\exp(\text{-}u_{en}\ R)] = 2(1.1)[1\text{-}\exp(\text{-}4.92\text{E-}5{*}905)] \sim 0.1$

- f is the submersion dose correction factor and has a value of ~ 0.1 or less and is applied to the submersion dose inside the reactor building

- f' the ratio of dose from a finite cloud to dose from a semi-infinite cloud given by the product of $u_{en}\ R$

- u_{en} = energy absorption coefficient in air for photons, for photons above 50 keV this value is < 4.92E-5 per cm from National Institute of Standards and Measurements, Tables of X-Ray Mass Attenuation Coefficients and Mass Energy-Absorption Coefficients from 1 keV to 20 MeV for Elements Z = 1 to 92 and 48 Additional Substances of Dosimetric Interest [Ref 30]

- R = effective radius of 905 cm based on the reactor building volume of 3.1E9 ml

- G = geometry correction factor of 2 for a sphere (4π geometry for personnel at an elevated location) vs. hemisphere (2π geometry for semi-infinite cloud affecting personnel on a lower level surface)

- k = ratio of mass energy absorption coefficients for tissue to air to convert to tissue dose having a value of ~ 1.1 for photon energies from 50 keV to several MeV

The estimated and measured free volumes are both above the FSAR value of 2.4 E9 ml [Ref 5] and TS value of 2.25 E9 ml [Ref 12]. Additional equipment, modifications, or experiments in the reactor building significantly affecting free air volume are not expected.

Therefore, the TS value of 2.25 E9 ml was used in this analysis.

Submersion Dose Correction f Value

Energy (MeV)	DRY AIR μen/ρ (cm2/g)	DRY AIR μen (1/cm)	DRY AIR μ/ρ (cm2/g)	1 - exp(-uenR)	f value
5.00E-02	4.10E-02	4.92E-05	2.08E-01	4.53E-02	9.97E-02
6.00E-02	3.04E-02	3.65E-05	1.88E-01	3.38E-02	7.44E-02
8.00E-02	2.41E-02	2.89E-05	1.66E-01	2.69E-02	5.91E-02
1.00E-01	2.33E-02	2.79E-05	1.54E-01	2.60E-02	5.71E-02
1.50E-01	2.50E-02	3.00E-05	1.36E-01	2.79E-02	6.13E-02
2.00E-01	2.67E-02	3.21E-05	1.23E-01	2.98E-02	6.55E-02
3.00E-01	2.87E-02	3.45E-05	1.07E-01	3.20E-02	7.04E-02
4.00E-01	2.95E-02	3.54E-05	9.55E-02	3.28E-02	7.22E-02
5.00E-01	2.97E-02	3.56E-05	8.71E-02	3.30E-02	7.26E-02
6.00E-01	2.95E-02	3.54E-05	8.06E-02	3.29E-02	7.23E-02
8.00E-01	2.88E-02	3.46E-05	7.07E-02	3.21E-02	7.06E-02
1.00E+00	2.79E-02	3.35E-05	6.36E-02	3.11E-02	6.84E-02
1.25E+00	2.67E-02	3.20E-05	5.69E-02	2.97E-02	6.54E-02
1.50E+00	2.55E-02	3.06E-05	5.18E-02	2.84E-02	6.25E-02
2.00E+00	2.35E-02	2.81E-05	4.45E-02	2.62E-02	5.76E-02
3.00E+00	2.06E-02	2.47E-05	3.58E-02	2.30E-02	5.06E-02
4.00E+00	1.87E-02	2.24E-05	3.08E-02	2.09E-02	4.61E-02
5.00E+00	1.74E-02	2.09E-05	2.75E-02	1.95E-02	4.29E-02
6.00E+00	1.65E-02	1.98E-05	2.52E-02	1.85E-02	4.06E-02
8.00E+00	1.53E-02	1.83E-05	2.23E-02	1.71E-02	3.76E-02
1.00E+01	1.45E-02	1.74E-05	2.05E-02	1.63E-02	3.58E-02
1.50E+01	1.35E-02	1.62E-05	1.81E-02	1.52E-02	3.34E-02
2.00E+01	1.31E-02	1.57E-05	1.71E-02	1.47E-02	3.24E-02

Volume

Measurements of the reactor building experimental area were made and give a total volume of 3.5 E9 ml. Free volume was measured to be 3.1 E9 ml by accounting for existing equipment and experiments:

Location	Cubic feet	Cubic cm
Reactor Bay	1.24E+05	3.52E+09
Loading Dock	7.38E+03	2.09E+08
Imaging and Positron	3.14E+03	8.90E+07
Reactor Shield Level 1	3.16E+03	8.94E+07
Pool level 1	5.87E+02	1.66E+07
Reactor Shield Level 2	1.95E+03	5.53E+07
Pool level 2	9.71E+02	2.75E+07
Reactor Shield Level 3	2.13E+02	6.04E+06
Pool level 3	3.31E+02	9.36E+06
Gross Free Volume	1.21E+05	3.43E+09
Misc Volume	1.21E+04	3.43E+08
Net Free Volume	1.09E+05	3.09E+09

EVACUATION AND PUBLIC EXPOSURE TIME

US NRC Regulatory Guide 2.2 [Ref 3] states that evacuation time is to be considered in the analysis for experiment failure. Evacuation time was measured from various locations inside the reactor building to the evacuation exit point for several individuals to be 1 minute or less following initiation of the reactor building evacuation signal.

Evacuation exit was at the northwest basement door as specified in the facility emergency plan and procedures. Also, an evacuation time of approximately 1 minute is calculated for an average walking pace of 3 mph [Ref 26,27] for 250 feet, which is the distance from the furthest location in the reactor building to the assembly point outside the reactor building. Measured times are in good agreement with estimated walking times.

REACTOR BUILDING EVACUATION TIMES

Location	Exit Point	Evacuation	Distance	Walking
		Time, s		Time, s
		(measured)		
Control Room	North West Door	58	250	57
Reactor pool top	North West Door	55	250	57
Mechanical Equipment Room	North West Door	45	200	45
Primary Piping Vault	North West Door	36	150	34
Loading dock	North West Door	54	225	51
Experiment/beam tubes (east)	North West Door	48	200	45
Experiment/beam tubes (west)	North West Door	25	100	23
Experiment/beam tubes (south)	North West Door	10	40	9
Experiment/beam tubes (north)	North West Door	38	175	40
Ventilation room	North West Door	10	40	9
Date:	19-May-16			

For personnel in the reactor building, an exposure time (T) of 3 minutes (180 seconds) is used based on the time needed for operator or detector action to activate building evacuation alarm and for personnel to physically exit the reactor building. Detector response time may take up to 60 seconds. Reactor Operator response may take up to 30 seconds. Total is 150 seconds (60+30+60) for evacuation to be complete. The evacuation time was increased to 3 minutes.

Exposure time (T) is taken as either 2 hours or 24 hours for dose assessment to members of the public. 2 hours allows sufficient time for detection and response by facility personnel to determine affected areas that need to be evacuated. 24 hours is sufficient time for the entire released activity to be vented from the reactor building (in excess of 10 air changes). A public exposure time of 24 hours is also associated with meeting emergency action levels given in the facility emergency plan [Ref 7,8].

DOSE RESULTS

For the U-235 reference case, the following doses were calculated using the equations and methods described in this analysis. For Public Exposure Time of 24 hours:

Nuclide	DAC uCi/ml	DAC uCi/ml	EC uCi/ml	Reactor Confinement TEDE rem	Thyroid Confinement CDE rem	Public Confinement TEDE rem
83mKr	1.00E-02		5.00E-05	1.07E-06		3.11E-07
85mKr	2.00E-05		1.00E-07	1.29E-03		5.07E-04
85Kr	1.00E-04		7.00E-07	5.69E-05		2.13E-05
87Kr	5.00E-06		2.00E-08	1.00E-02		3.02E-03
88Kr	2.00E-06		9.00E-09	3.49E-02		1.33E-02
89Kr	1E-7		1E-9	6.41E-01		7.59E-03
90Kr	1E-7		1E-9	2.44E-01		1.46E-03
91Kr	1E-7		1E-9	4.55E-02		2.66E-04
92Kr	1E-7		1E-9	4.78E-03		2.80E-05
93Kr	1E-7		1E-9	9.54E-04		5.59E-06
94Kr	1E-7		1E-9	2.76E-05		1.62E-07
95Kr	1E-7		1E-9	1.33E-05		7.78E-08
96Kr	1E-7		1E-9	3.88E-06		2.27E-08
97Kr	1E-7		1E-9	4.91E-09		2.88E-11
131mXe	4.00E-04		2.00E-06	1.96E-06		1.02E-06
133mXe	1.00E-04		6.00E-07	3.80E-05		1.61E-05
133Xe	1.00E-04		5.00E-07	1.34E-03		6.90E-04
135mXe	9.00E-06		4.00E-08	2.35E-03		2.11E-04
135Xe	1.00E-05		7.00E-08	1.30E-02		4.17E-03
137Xe	1E-7		1E-9	9.38E-01		1.27E-02
138Xe	4.00E-06		2.00E-08	2.89E-02		2.16E-03
139Xe	1E-7		1E-9	2.96E-01		1.81E-03
140Xe	1E-7		1E-9	7.54E-02		4.41E-04
141Xe	1E-7		1E-9	3.25E-03		1.91E-05
142Xe	1E-7		1E-9	9.20E-04		5.39E-06
143Xe	1E-7		1E-9	2.18E-05		1.28E-07
144Xe	1E-7		1E-9	1.83E-05		1.07E-07
145Xe	1E-7		1E-9	7.32E-08		4.29E-10
131I	8.33E-08	2.00E-08	2.00E-10	7.08E+00	2.95E+02	7.65E-02
132I	4.17E-06	3.00E-06	2.00E-08	2.08E-01	2.89E+00	6.86E-04
133I	3.75E-07	1.00E-07	1.00E-09	3.56E+00	1.33E+02	3.25E-02
134I	2.00E-05	2.35E-05	6.00E-08	7.61E-02	6.47E-01	2.46E-04
135I	1.67E-06	7.00E-07	6.00E-09	7.49E-01	1.78E+01	4.42E-03
83Br	3.00E-05		9.00E-08	3.61E-03		1.94E-05
84Br	2.00E-05		8.00E-08	9.41E-03		1.64E-05
84mBr	1E-7		1E-9	3.64E-02		6.90E-06
85Br	1E-7		1E-9	1.86E+00		2.07E-04
86Br	1E-7		1E-9	1.43E+00		9.34E-05
87Br	1E-7		1E-9	1.66E+00		1.09E-04
88Br	1E-7		1E-9	4.81E-01		2.82E-05
89Br	1E-7		1E-9	8.25E-02		4.83E-06
90Br	1E-7		1E-9	1.85E-02		1.08E-06
91Br	1E-7		1E-9	1.79E-03		1.05E-07
92Br	1E-7		1E-9	1.40E-04		8.20E-09
93Br	1E-7		1E-9	2.65E-05		1.55E-09
96Br	1E-7		1E-9	4.98E-10		2.92E-14
Rb88	3.00E-05			7.69E-05		
Rb89	6.00E-05			3.22E-05		
Rb90	1E-7			1.41E-02		
Rb91	1E-7			1.57E-03		
Rb92	1E-7			3.52E-05		
Rb93	1E-7			4.93E-06		
Rb94	1E-7			2.33E-08		
Rb95	1E-7			4.15E-08		
Rb96	1E-7			1.25E-09		
Cs138	2.00E-05			1.13E-04		
Cs139	1E-7			7.30E-03		
Cs140	1E-7			3.38E-03		
Cs141	1E-7			2.23E-05		
Cs142	1E-7			4.50E-06		
Cs143	1E-7			2.63E-08		
Cs144	1E-7	Reactor Noble Gas DDE =	2.34E+00	8.77E-08	4.52E+02	= Reactor Thyroid TODE
Cs145	1E-7	Public Noble Gas DDE =	2.34E+01	5.52E-11	4.73E+02	= Public Thyroid TODE
		Reactor TEDE =		1.96E+01	Public TEDE =	1.63E-01

For the U-235 reference case, the following doses were calculated using the equations and methods described in this analysis. For Public Exposure Time of 2 hours:

Nuclide	DAC uCi/ml	DAC uCi/ml	EC uCi/ml	Reactor Confinement TEDE rem	Thyroid Confinement CDE rem	Public Confinement TEDE rem
83mKr	1.00E-02		5.00E-05	1.07E-06		8.50E-07
85mKr	2.00E-05		1.00E-07	1.29E-03		1.20E-03
85Kr	1.00E-04		7.00E-07	5.69E-05		4.28E-05
87Kr	5.00E-06		2.00E-08	1.00E-02		8.82E-03
88Kr	2.00E-06		9.00E-09	3.49E-02		3.37E-02
89Kr	1E-7		1E-9	6.41E-01		2.56E-02
90Kr	1E-7		1E-9	2.44E-01		4.93E-03
91Kr	1E-7		1E-9	4.55E-02		9.00E-04
92Kr	1E-7		1E-9	4.78E-03		9.45E-05
93Kr	1E-7		1E-9	9.54E-04		1.89E-05
94Kr	1E-7		1E-9	2.76E-05		5.47E-07
95Kr	1E-7		1E-9	1.33E-05		2.63E-07
96Kr	1E-7		1E-9	3.88E-06		7.68E-08
97Kr	1E-7		1E-9	4.91E-09		9.72E-11
131mXe	4.00E-04		2.00E-06	1.96E-06		2.06E-06
133mXe	1.00E-04		6.00E-07	3.80E-05		3.30E-05
133Xe	1.00E-04		5.00E-07	1.34E-03		1.40E-03
135mXe	9.00E-06		4.00E-08	2.35E-03		7.13E-04
135Xe	1.00E-05		7.00E-08	1.30E-02		9.20E-03
137Xe	1E-7		1E-9	9.38E-01		4.29E-02
138Xe	4.00E-06		2.00E-08	2.89E-02		7.30E-03
139Xe	1E-7		1E-9	2.96E-01		6.12E-03
140Xe	1E-7		1E-9	7.54E-02		1.49E-03
141Xe	1E-7		1E-9	3.25E-03		6.44E-05
142Xe	1E-7		1E-9	9.20E-04		1.82E-05
143Xe	1E-7		1E-9	2.18E-05		4.32E-07
144Xe	1E-7		1E-9	1.83E-05		3.61E-07
145Xe	1E-7		1E-9	7.32E-08		1.45E-09
131I	8.33E-08	2.00E-08	2.00E-10	7.08E+00	2.95E+02	1.55E-01
132I	4.17E-06	3.00E-06	2.00E-08	2.08E-01	2.89E+00	1.80E-03
133I	3.75E-07	1.00E-07	1.00E-09	3.56E+00	1.33E+02	6.83E-02
134I	2.00E-05	2.35E-05	6.00E-08	7.61E-02	6.47E-01	7.63E-04
135I	1.67E-06	7.00E-07	6.00E-09	7.49E-01	1.78E+01	1.00E-02
83Br	3.00E-05		9.00E-08	3.61E-03		5.05E-05
84Br	2.00E-05		8.00E-08	9.41E-03		5.36E-05
84mBr	1E-7		1E-9	3.64E-02		2.33E-05
85Br	1E-7		1E-9	1.86E+00		6.99E-04
86Br	1E-7		1E-9	1.43E+00		3.16E-04
87Br	1E-7		1E-9	1.66E+00		3.67E-04
88Br	1E-7		1E-9	4.81E-01		9.52E-05
89Br	1E-7		1E-9	8.25E-02		1.63E-05
90Br	1E-7		1E-9	1.85E-02		3.66E-06
91Br	1E-7		1E-9	1.79E-03		3.53E-07
92Br	1E-7		1E-9	1.40E-04		2.77E-08
93Br	1E-7		1E-9	2.65E-05		5.24E-09
96Br	1E-7		1E-9	4.98E-10		9.85E-14
Rb88	3.00E-05			7.69E-05		
Rb89	6.00E-05			3.22E-05		
Rb90	1E-7			1.41E-02		
Rb91	1E-7			1.57E-03		
Rb92	1E-7			3.52E-05		
Rb93	1E-7			4.93E-06		
Rb94	1E-7			2.33E-08		
Rb95	1E-7			4.15E-08		
Rb96	1E-7			1.25E-09		
Cs138	2.00E-05			1.13E-04		
Cs139	1E-7			7.30E-03		
Cs140	1E-7			3.38E-03		
Cs141	1E-7			2.23E-05		
Cs142	1E-7			4.50E-06		
Cs143	1E-7			2.63E-08		
Cs144	1E-7	Reactor Noble Gas DDE =	2.34E+00	8.77E-08	4.52E+02	= Reactor Thyroid TODE
Cs145	1E-7	Public Noble Gas DDE =	2.34E+01	5.52E-11	4.73E+02	= Public Thyroid TODE
		Reactor TEDE =		1.96E+01	Public TEDE =	3.82E-01

For the reference case of any fissionable material, the following doses were calculated using the equations and methods described in this analysis. For Public Exposure Time of 24 hours:

Nuclide	DAC uCi/ml	DAC uCi/ml	EC uCi/ml	Reactor Confinement TEDE rem	Thyroid Confinement CDE rem	Public Confinement TEDE rem
83mKr	1.00E-02		5.00E-05	8.18E-05		2.37E-05
85mKr	2.00E-05		1.00E-07	4.37E-02		1.71E-02
85Kr	1.00E-04		7.00E-07	8.89E-03		3.32E-03
87Kr	5.00E-06		2.00E-08	2.40E-01		7.21E-02
88Kr	2.00E-06		9.00E-09	6.44E-01		2.45E-01
89Kr	1E-7		1E-9	1.05E+01		1.24E-01
90Kr	1E-7		1E-9	2.92E+00		1.75E-02
91Kr	1E-7		1E-9	4.91E-01		2.88E-03
92Kr	1E-7		1E-9	5.66E-02		3.31E-04
93Kr	1E-7		1E-9	1.47E-02		8.59E-05
94Kr	1E-7		1E-9	9.13E-04		5.35E-06
95Kr	1E-7		1E-9	6.15E-04		3.60E-06
96Kr	1E-7		1E-9	4.09E-05		2.40E-07
97Kr	1E-7		1E-9	1.27E-06		7.44E-09
131mXe	4.00E-04		2.00E-06	1.82E-05		9.49E-06
133mXe	1.00E-04		6.00E-07	3.38E-04		1.43E-04
133Xe	1.00E-04		5.00E-07	1.19E-02		6.17E-03
135mXe	9.00E-06		4.00E-08	2.14E-02		1.92E-03
135Xe	1.00E-05		7.00E-08	1.30E-01		4.18E-02
137Xe	1E-7		1E-9	8.93E+00		1.21E-01
138Xe	4.00E-06		2.00E-08	2.77E-01		2.07E-02
139Xe	1E-7		1E-9	3.61E+00		2.21E-02
140Xe	1E-7		1E-9	8.97E-01		5.25E-03
141Xe	1E-7		1E-9	6.23E-02		3.65E-04
142Xe	1E-7		1E-9	1.94E-02		1.14E-04
143Xe	1E-7		1E-9	1.27E-03		7.45E-06
144Xe	1E-7		1E-9	1.27E-03		7.41E-06
145Xe	1E-7		1E-9	1.66E-05		9.73E-08
131I	8.33E-08	2.00E-08	2.00E-10	8.03E+01	3.35E+03	8.69E-01
132I	4.17E-06	3.00E-06	2.00E-08	2.19E+00	3.04E+01	7.21E-03
133I	3.75E-07	1.00E-07	1.00E-09	3.18E+01	1.19E+03	2.91E-01
134I	2.00E-05	2.35E-05	6.00E-08	6.56E-01	5.59E+00	2.13E-03
135I	1.67E-06	7.00E-07	6.00E-09	7.47E+00	1.78E+02	4.41E-02
83Br	3.00E-05		9.00E-08	2.73E-01		1.46E-03
84Br	2.00E-05		8.00E-08	6.66E-01		1.16E-03
84mBr	1E-7		1E-9	4.85E+00		9.20E-04
85Br	1E-7		1E-9	6.23E+01		6.94E-03
86Br	1E-7		1E-9	4.07E+01		2.66E-03
87Br	1E-7		1E-9	3.58E+01		2.34E-03
88Br	1E-7		1E-9	7.81E+00		4.58E-04
89Br	1E-7		1E-9	1.30E+00		7.59E-05
90Br	1E-7		1E-9	2.66E-01		1.56E-05
91Br	1E-7		1E-9	2.97E-02		1.74E-06
92Br	1E-7		1E-9	3.84E-03		2.25E-07
93Br	1E-7		1E-9	2.86E-04		1.68E-08
96Br	1E-7		1E-9	2.57E-08		1.51E-12
Rb88	3.00E-05			1.42E-03		
Rb89	6.00E-05			5.26E-04		
Rb90	1E-7			1.68E-01		
Rb91	1E-7			1.69E-02		
Rb92	1E-7			4.17E-04		
Rb93	1E-7			7.58E-05		
Rb94	1E-7			7.68E-07		
Rb95	1E-7			1.92E-06		
Rb96	1E-7			1.31E-08		
Cs138	2.00E-05			1.08E-03		
Cs139	1E-7			8.90E-02		
Cs140	1E-7			4.02E-02		
Cs141	1E-7			4.26E-04		
Cs142	1E-7			9.49E-05		
Cs143	1E-7			1.53E-06		
Cs144	1E-7	Reactor Noble Gas DDE =	2.89E+01	6.09E-06	4.78E+03	= Reactor Thyroid TODE
Cs145	1E-7	Public Noble Gas DDE =	2.89E+02	1.25E-08	5.04E+03	= Public Thyroid TODE
		Reactor TEDE =		3.06E+02	Public TEDE =	1.93E+00

For the reference case of any fissionable material, the following doses were calculated using the equations and methods described in this analysis. For Public Exposure Time of 2 hours:

Nuclide	DAC uCi/ml	DAC uCi/ml	EC uCi/ml	Reactor Confinement TEDE rem	Thyroid Confinement CDE rem	Public Confinement TEDE rem
83mKr	1.00E-02		5.00E-05	8.18E-05		6.47E-05
85mKr	2.00E-05		1.00E-07	4.37E-02		4.06E-02
85Kr	1.00E-04		7.00E-07	8.89E-03		6.68E-03
87Kr	5.00E-06		2.00E-08	2.40E-01		2.10E-01
88Kr	2.00E-06		9.00E-09	6.44E-01		6.22E-01
89Kr	1E-7		1E-9	1.05E+01		4.19E-01
90Kr	1E-7		1E-9	2.92E+00		5.90E-02
91Kr	1E-7		1E-9	4.91E-01		9.72E-03
92Kr	1E-7		1E-9	5.66E-02		1.12E-03
93Kr	1E-7		1E-9	1.47E-02		2.90E-04
94Kr	1E-7		1E-9	9.13E-04		1.81E-05
95Kr	1E-7		1E-9	6.15E-04		1.22E-05
96Kr	1E-7		1E-9	4.09E-05		8.09E-07
97Kr	1E-7		1E-9	1.27E-06		2.51E-08
131mXe	4.00E-04		2.00E-06	1.82E-05		1.92E-05
133mXe	1.00E-04		6.00E-07	3.38E-04		2.93E-04
133Xe	1.00E-04		5.00E-07	1.19E-02		1.25E-02
135mXe	9.00E-06		4.00E-08	2.14E-02		6.49E-03
135Xe	1.00E-05		7.00E-08	1.30E-01		9.21E-02
137Xe	1E-7		1E-9	8.93E+00		4.08E-01
138Xe	4.00E-06		2.00E-08	2.77E-01		7.00E-02
139Xe	1E-7		1E-9	3.61E+00		7.46E-02
140Xe	1E-7		1E-9	8.97E-01		1.77E-02
141Xe	1E-7		1E-9	6.23E-02		1.23E-03
142Xe	1E-7		1E-9	1.94E-02		3.84E-04
143Xe	1E-7		1E-9	1.27E-03		2.52E-05
144Xe	1E-7		1E-9	1.27E-03		2.50E-05
145Xe	1E-7		1E-9	1.66E-05		3.29E-07
131I	8.33E-08	2.00E-08	2.00E-10	8.03E+01	3.35E+03	1.76E+00
132I	4.17E-06	3.00E-06	2.00E-08	2.19E+00	3.04E+01	1.90E-02
133I	3.75E-07	1.00E-07	1.00E-09	3.18E+01	1.19E+03	6.11E-01
134I	2.00E-05	2.35E-05	6.00E-08	6.56E-01	5.59E+00	6.58E-03
135I	1.67E-06	7.00E-07	6.00E-09	7.47E+00	1.78E+02	1.00E-01
83Br	3.00E-05		9.00E-08	2.73E-01		3.82E-03
84Br	2.00E-05		8.00E-08	6.66E-01		3.79E-03
84mBr	1E-7		1E-9	4.85E+00		3.11E-03
85Br	1E-7		1E-9	6.23E+01		2.34E-02
86Br	1E-7		1E-9	4.07E+01		8.98E-03
87Br	1E-7		1E-9	3.58E+01		7.91E-03
88Br	1E-7		1E-9	7.81E+00		1.55E-03
89Br	1E-7		1E-9	1.30E+00		2.56E-04
90Br	1E-7		1E-9	2.66E-01		5.25E-05
91Br	1E-7		1E-9	2.97E-02		5.88E-06
92Br	1E-7		1E-9	3.84E-03		7.60E-07
93Br	1E-7		1E-9	2.86E-04		5.66E-08
96Br	1E-7		1E-9	2.57E-08		5.08E-12
Rb88	3.00E-05			1.42E-03		
Rb89	6.00E-05			5.26E-04		
Rb90	1E-7			1.68E-01		
Rb91	1E-7			1.69E-02		
Rb92	1E-7			4.17E-04		
Rb93	1E-7			7.58E-05		
Rb94	1E-7			7.68E-07		
Rb95	1E-7			1.92E-06		
Rb96	1E-7			1.31E-08		
Cs138	2.00E-05			1.08E-03		
Cs139	1E-7			8.90E-02		
Cs140	1E-7			4.02E-02		
Cs141	1E-7			4.26E-04		
Cs142	1E-7			9.49E-05		
Cs143	1E-7			1.53E-06		
Cs144	1E-7	Reactor Noble Gas DDE =	2.89E+01	6.09E-06	4.78E+03	= Reactor Thyroid TODE
Cs145	1E-7	Public Noble Gas DDE =	2.89E+02	1.25E-08	5.04E+03	= Public Thyroid TODE
		Reactor TEDE =		3.06E+02	Public TEDE =	4.60E+00

EXPERIMENT LIMITS

The postulated dose from an inadvertent release depends on the activity present. Calculations were performed for U-235 and for other fissionable materials for reference conditions to produce saturation activities.

Reference fluence rate was 7 E12 $cm^{-2}s^{-1}$ for thermal fission and 2 E12 $cm^{-2}s^{-1}$ for non-thermal fission. Reference mass was 1 gram throughout the irradiation time. No decay time is assumed since the release may occur during the experiment.

The limiting fission rate, f/s^{Limit}, was calculated as follows:

$$f/s^{Limit} = \frac{Dose \quad Limit}{Total \quad reference \quad dose \quad per \quad fission \quad rate} \tag{EQ 25}$$

where, Dose limits are set at the regulatory limits:

- o 5 rem for TEDE inside the reactor facility
- o 50 rem TODE to the thyroid inside the reactor facility
- o 0.1 rem TEDE in unrestricted areas (public areas outside the reactor facility)

Total reference dose per unit fission rate (in rem per f/s) was determined by summing the TEDE or TODE for all radionuclides released and dividing the TEDE or TODE by the total fission rate from thermal and non-thermal fission.

For TEDE, the submersion DDE was added to the inhalation CEDE. For TODE to the thyroid, the submersion DDE was added to the thyroid inhalation CDE.

For TODE to the thyroid in public areas outside the reactor building, the submersion dose correction factor is not used; i.e. f = 1, not 0.1.

NON-FUELED EXPERIMENTS

Non-fueled experiments are analyzed as described previously except that the reactor building is operated in the normal mode and the annual public TEDE limit of 0.1 rem is met either inside or outside the reactor building.

Given the magnitude of the activity released, the reactor facility radiation monitoring system would detect such a release and the evacuation and confinement system would activate. The evacuation time of 3 minutes is applicable. Public exposure times of 2 hours and 24 hours were analyzed.

For a release time of 24 hours, the following limiting fission rate and non-fueled experiment fission rate are calculated using EQ 25 for U-235:

FUELED EXPERIMENT DOSE SUMMARY AND RESULTS				
	Confinement Effective rem	Confinement Thyroid rem	Confinement Public Effective rem	Public Hours
Noble Gas	2.34E+00		4.84E-02	2.40E+01
Iodines	1.17E+01	4.52E+02	1.14E-01	
Bromines	5.59E+00		4.86E-04	
Rb,Cs	2.66E-02		0.00E+00	
Total dose =	1.96E+01	4.52E+02	1.63E-01	
rem dose per f/s=	1.46E-12	3.37E-11	1.22E-14	LIMIT
Dose limit, rem =	5	50	0.1	
Total fission rate =	3.42E+12	1.48E+12	8.22E+12	**1.48E+12 f/s**

NON-FUELED DOSE SUMMARY AND RESULTS				
	Reactor Building	Public Areas	Public Areas	
Total dose =	1.96E+01	4.73E+02	1.63E-01	
rem dose per f/s=	1.46E-12	3.53E-11	1.22E-14	LIMIT
Dose limit, rem =	0.1		0.1	
Total fission rate =	6.84E+10		8.22E+12	**6.84E+10 f/s**

For a release time of 2 hours, the following limiting fission rate and non-fueled experiment fission rate are calculated using EQ 25 for U-235:

FUELED EXPERIMENT DOSE SUMMARY AND RESULTS				
	Confinement Effective rem	Confinement Thyroid rem	Confinement Public Effective rem	Public Hours
Noble Gas	2.34E+00		1.45E-01	2.00E+00
Iodines	1.17E+01	4.52E+02	2.36E-01	
Bromines	5.59E+00		1.63E-03	
Rb,Cs	2.66E-02		0.00E+00	
Total dose =	1.96E+01	4.52E+02	3.82E-01	
rem dose per f/s=	1.46E-12	3.37E-11	2.85E-14	LIMIT
Dose limit, rem =	5	50	0.1	
Total fission rate =	3.42E+12	1.48E+12	3.51E+12	**1.48E+12 f/s**

NON-FUELED DOSE SUMMARY AND RESULTS				
	Reactor Building	Public Areas	Public Areas	
Total dose =	1.96E+01	4.73E+02	3.82E-01	
rem dose per f/s=	1.46E-12	3.53E-11	2.85E-14	LIMIT
Dose limit, rem =	0.1		0.1	
Total fission rate =	6.84E+10		3.51E+12	**6.84E+10 f/s**

The limiting fission rate of 1.5 E12 f/s for either public exposure time case are based on the thyroid TODE inside the reactor building. The non-fueled experiment limit of 6.8 E10 f/s for either public exposure time case are based on the TEDE inside the reactor building.

The thyroid TODE is estimated to be 2.3 rem (6.84 E10 / 1.48 E12 * 50 rem) at this non-fueled experiment f/s limit inside the reactor building.

For a release time of 24 hours, the following limiting fission rate and non-fueled experiment fission rate are calculated using EQ 25 for any fissionable material:

FUELED EXPERIMENT DOSE SUMMARY AND RESULTS				
	Confinement Effective rem	Confinement Thyroid rem	Confinement Public Effective rem	Public Hours
Noble Gas	2.89E+01		7.02E-01	2.40E+01
Iodines	1.22E+02	4.78E+03	1.21E+00	
Bromines	1.54E+02		1.60E-02	
Rb,Cs	3.19E-01		0.00E+00	
Total dose =	3.06E+02	4.78E+03	1.93E+00	
rem dose per f/s=	2.66E-12	4.17E-11	1.68E-14	**LIMIT**
Dose limit, rem =	5	50	0.1	
Total fission rate =	1.88E+12	1.20E+12	5.95E+12	**1.20E+12 f/s**
NON-FUELED DOSE SUMMARY AND RESULTS				
	Reactor Building	**Public Areas**	**Public Areas**	
Total dose =	3.06E+02	5.04E+03	1.93E+00	
rem dose per f/s=	2.66E-12	4.39E-11	1.68E-14	**LIMIT**
Dose limit, rem =	0.1		0.1	
Total fission rate =	3.76E+10		5.95E+12	**3.76E+10 f/s**

For a release time of 2 hours, the following limiting fission rate and non-fueled experiment fission rate are calculated using EQ 25 for any fissionable material:

FUELED EXPERIMENT DOSE SUMMARY AND RESULTS				
	Confinement Effective rem	Confinement Thyroid rem	Confinement Public Effective rem	Public Hours
Noble Gas	2.89E+01		2.05E+00	2.00E+00
Iodines	1.22E+02	4.78E+03	2.49E+00	
Bromines	1.54E+02		5.29E-02	
Rb,Cs	3.19E-01		0.00E+00	
Total dose =	3.06E+02	4.78E+03	4.60E+00	
rem dose per f/s=	2.66E-12	4.17E-11	4.01E-14	**LIMIT**
Dose limit, rem =	5	50	0.1	
Total fission rate =	1.88E+12	1.20E+12	2.50E+12	**1.20E+12 f/s**
NON-FUELED DOSE SUMMARY AND RESULTS				
	Reactor Building	**Public Areas**	**Public Areas**	
Total dose =	3.06E+02	5.04E+03	4.60E+00	
rem dose per f/s=	2.66E-12	4.39E-11	4.01E-14	**LIMIT**
Dose limit, rem =	0.1		0.1	
Total fission rate =	3.76E+10		2.50E+12	**3.76E+10 f/s**

The limiting fission rate of 1.2 E12 f/s for either public exposure time case are based on the thyroid TODE inside the reactor building. The non-fueled experiment limit of 3.7 E10 f/s for either public exposure time case are based on the TEDE inside the reactor building.

The thyroid TODE is estimated to be 1.6 rem (3.76 E10/1.2 E12 * 50 rem) at this non-fueled experiment f/s limit inside the reactor building.

Similar results were observed for thermal fission only or non-thermal fission only for any fissionable material:

Fission	Fueled Experiment f/s	Non-Fueled Experiment f/s
Thermal	1.21 E12	3.77 E10
Non-thermal	1.15 E12	3.72 E10
Combined	1.21 E12	3.76 E10

Limiting Fission Rates

The limiting fission rates are rounded off to 1.2 E12 f/s for fueled experiments and 3.7 E10 f/s for non-fueled experiments. These limiting fission rates apply to all fissionable material present.

Emergency Action Levels (EAL): [Ref 7,8]

EAL are established for radiation dose to the members of the public over 24 hours. TEDE and TODE to the thyroid at the proposed f/s limit and non-fueled experiment limits are made using the Total Reference Dose per unit fission rate and limiting fission rates as previously reported for any fissionable material:

At the fueled experiment f/s limit for any fissionable material:

TEDE: 2.0 E-2 rem = (1.2E12 f/s)(1.68E-14 rem per f/s)
TODE: 6.4 E-3 rem = (1.2E12 f/s)(4.17E-11 rem per f/s)(4.53E-3 s/m^3)(0.283 m^3/s)(1-0.9)

At the non-fueled experiment f/s limit for any fissionable material:

TEDE: 6.3 E-4 rem = (3.7E10 f/s)(1.68E-14 rem per f/s)
TODE: 2.1 E-4 rem = (3.7E10 f/s)(4.39E-11 rem per f/s)(4.53E-3 s/m^3)(0.283 m^3/s)(1-0.9)

The fission rate for a fueled experiment and non-fueled experiment are compared to EAL classifications based on radiation dose below:

EAL	DDE rem	CEDE rem	CDE rem	TEDE rem	TODE rem	EAL Fraction TEDE	EAL Fraction TODE
Notification of Unusual Event	0.015	0.015		0.03		6.7E-01	
Alert	0.075	0.075	0.1	0.15	0.175	1.3E-01	3.7E-02
Site Area Emergency	0.375	0.375	0.5	0.75	0.875	2.7E-02	7.3E-03
General Area Emergency	0.5		5	0.5	5	4.0E-02	1.3E-03
		Fueled f/s limit		2.0E-02	6.4E-03		
		Non fueled f/s limit		6.3E-04	2.1E-04		

At the fueled experiment fission rate limit of 1.2E12 f/s and non-fueled experiment limit of 3.7 E10 f/s, the 24 hour public TEDE and thyroid TODE are below all EAL classifications.

Variance with Fluence Rate:

Mass, or N, varies inversely with φ (fluence rate) to maintain the same limiting fission rate. The mass has practical limits based on other experiment conditions, e.g. radiation dose rates from the encapsulated sample, handling, storage, disposal, and license limits. The mass and conditions are included in the experiment review required by TS 3.8.

From the limiting mass, the limiting fission rate is calculated as follows:

$$[f/s]^{Limit} = \sigma\varphi N \qquad \boxed{\text{EQ 26}}$$

Fission Rate Variance with Time

In this analysis, the fission rate limit giving the occupational or public annual dose limit was determined for any fissionable material. Depletion of the initial fissionable material and the presence of other fissionable material by production and decay were neglected by assumption. Given this assumption, the fission rate remains constant. However, the fission rate may decrease or increase with time if there is a significant change from decay and reaction by fission or activation.

If the fission rate decreases with time, the saturation activities would eventually decrease as well. An increase of the fission rate with time is of concern for planning fueled experiments.

In general, the fission rate would increase with time for the following conditions:

- Fission cross-section of the initial fissionable material is less than the activation cross-section
- Activation product has a large fission cross-section
- Irradiation time is long

These conditions would be evaluated as part of the fueled experiment review as specified in TS 3.8. All fissionable materials present initially and other fissionable materials produced by activation and/or decay are to be identified in the experiment review process. If the fission rate increases with time, either the mass or fluence rate would be adjusted to keep the fission rate within the fission rate limit of 1.2 E12 f/s over the planned duration of the fueled experiment.

Experiment Limits for Mixtures:

The maximum saturation activity and time integrated exposure data apply to any mixture. Results for individual fissionable materials would be at or less than those for the worst case (maximum conditions).

Fission rates from each fissionable material in the mixture are determined using EQ 26 and then summed to give the total fission rate for the sample. Corrections to N are made for the actual mass present, e.g. by using reported enrichment and assay data. Corrections are made to N to account for the fission rate variance with time as described above. The mass of the sample allowed is adjusted so that the limiting fission rate of 1.2 E12 f/s is not exceeded at any time over the planned duration of the experiment.

POSSESSION LIMITS

Possession limits given in the proposed facility license for fueled experiments are changed to specifically indicate fueled experiment inventory and to be within 10 CFR Part 37 Category 2 limits [Ref 28]. 10 CFR Part 37 Category 2 nuclides, activity limits, and associated mass limits are as follows:

Radionuclide	10 CFR Part 37 Category 2 Activity Limit, Ci	Mass Limit, g
Pu238	1.62E+01	9.53E-01
Pu239	1.62E+01	2.61E+02
Am241	1.62E+01	4.67E+02
Cm244	1.35E+01	1.67E-01
Cf252	5.40E+00	1.00E-02
Sr90	3.70E+02	1.93E+00
Cs137	2.70E+01	3.10E-01
Pm147	1.08E+04	1.16E+01
Gd153	2.70E+02	7.71E-02
Tm170	5.40E+03	9.00E+00
Yt169	8.10E+01	3.38E-02
Ir192	2.16E+01	2.53E-03
Ra226	1.08E+01	1.08E+01
Se75	5.40E+01	3.60E-03

Fission product inventory for fueled experiments for 10 years of continuous operation at a fission rate of 1.2 E12 f/s gives approximately 2% of the 10 CFR Part 37 Category 2 activity limits.

Nuclide	Half-life, s	Decay Constant, 1/s	Thermal Fission Yield %	Non-Thermal Fission Yield %	Activity Ci	10 CFR Part 37 Category 2 Limit, Ci	Fraction of Limit
Sr90	9.07E+08	7.65E-10	8.39	8.28	5.81E-01	3.70E+02	1.57E-03
Cs137	9.47E+08	7.32E-10	6.73	6.18	4.41E-01	2.70E+01	1.63E-02
Pm147	8.26E+07	8.39E-09	3.52	3.28	1.04E+00	1.08E+04	9.66E-05
Operation time =	1.00E+01	years					1.80E-02

Sr-90, Cs-137, and Pm-147 are listed above. Other fission products listed in 10 CFR Part 37 are produced in insignificant quantities due to a low cumulative fission yield.

Continuous operation for 10 years for a fueled experiment is not likely. A sample may be used several times over the course of several years. 10 years of continuous operation is equivalent to a total thermal fluence of 2.21 E21 cm^{-2} and a total non-thermal fluence of thermal fluence of 6.3 E20 cm^{-2}.

Therefore, the buildup of fission products that are listed in 10 CFR Part 37 from fueled experiments is estimated to be minor.

The proposed limits for special nuclear material requested are a fraction of the 10 CFR Part 37 Category 2 limits for both individual and the collective group of all material.

Possession limits are established based experimental needs and 10 CFR Part 37 limits as follows:

- Approximately half of the limit given for Category 2 in 10 CFR Part 37 for individual nuclides
- Sum of the fractions for all nuclides are not to exceed 0.9 of the limits listed in Category 2 of 10 CFR Part 37
- 100 g individually or 200 g total for all other special nuclear material not listed in 10 CFR Part 37, e.g. isotopes of U, other Pu, Np, Am, Cm

Fissionable Material	Mass Limit (g)	Sum of Fractions
Pu238 Pu239 Am241 Cm244 Cf252	0.47 100 2.2 0.08 0.005	Not to exceed 0.9 for all
Special Nuclear Material not listed above	100 each or 200 total	

TS 5.3 requirements for fueled experiments in storage shall be met, as applicable. Calculations and measurements made for reactor fuel are used for fueled experiment storage. These are documented using facility procedures to verify fueled experiments are stored in a configuration to keep k_{eff} no greater than 0.9.

Storage facilities are reviewed under TS 3.8, 10 CFR Part 50.59 for design changes, 10 CFR 50.54(p) for security, 10 CFR 50.54(q) for emergency planning, and 10 CFR Part 20 for radiation protection. If a storage facility requires a license amendment, then approval by the NRC is needed prior to implementing the change.

Fueled experiment materials are inventoried and accounted for as required by 10 CFR Part 70, as applicable, the university broad scope license, and facility procedures.

CONCLUSIONS

TS 3.8 on limitations and conditions for fueled experiments are revised based on satisfying the following criteria in this analysis:

- Radiation dose from an experiment failure does not exceed the occupational annual limits inside the reactor building or the annual limits to members of the public outside the reactor building given in 10 CFR Part 20.
- Non-fueled experiments are based on an experiment failure that does not exceed the public TEDE annual dose limit given in 10 CFR Part 20, either inside or outside the reactor building.
- An experiment failure does not exceed the emergency action level for an unusual event as defined in the facility emergency plan.
- An experiment failure does not exceed allowable limits and therefore does not meet the requirement for a reportable event as defined in TS 1.2.24.b.
- Fission product activity produced by a fueled experiment is minor relative to the quantity of concern limits given in 10 CFR Part 37 for Category 2.

Results from the analysis are summarized below:

- The limiting total fission rate was calculated to be 1.2 E12 f/s. The thyroid dose (TODE) inside the reactor building was the limiting dose used to determine the fission rate.
- A total fission rate of 3.7 E10 f/s was calculated to result in a TEDE of 0.1 rem to occupants inside the reactor building or members of the public outside the reactor building. Experiments less than 3.7 E10 f/s or a TEDE of 0.1 rem are not considered to be fueled experiments.
- Fissionable material, either individually or in mixtures, that meet experiment and license conditions may be used in a fueled experiment.
- The fissionable material mass and experiment fluence rates may be adjusted such that the fission rate limitation is met.

TS 1.2.9e was revised to provide masses of fissionable materials that do not exceed the annual public dose limit and to exclude self-encapsulated items such as detectors, foils, wire, sealed sources, and fuel with cladding.

Proposed possession limits for the reactor facility license do not exceed quantity of concern limits given in 10 CFR Part 37 for Category 2. Materials shall be stored as required by TS and the facility security plan and radiation protection program.

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