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27 August 2018

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 Specifications 3.8 and 1.2.9 e for fueled experiments. This request is based on a revised analysis and replaces the previous submittal made on 21 August 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 27 August 2018.

Sincerely,

Ayman Hawari

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Enclosures: Technical Specification Amendment
Attachment 1: Fueled Experiment Analysis

cc: Duane Hardesty, US NRC

Executive Summary

An amendment to the Technical Specifications and the reactor license is requested based on planned experimental needs. The requested changes are as follows:

1) Technical Specification (TS) 1.2.9.e for the definition of a fueled experiment
2) Technical Specification (TS) 3.8 for limiting conditions for operations for fueled experiments
3) R-120 reactor license Section 2.B.(2) regarding possession limits for fissionable materials to be used in fueled experiments

TS 1.2.9.e defines fueled experiments, and is revised to define the type and quantity of fissionable material that is a fueled experiment. Threshold limits on fission rate and total number of fissions are established for experiments containing uranium which are based on limiting the radiation dose from a potential release to one percent (1%) of the annual public dose limit given in 10 CFR Part 20, i.e. a Total Effective Dose-Equivalent of 0.001 rem. Experiments involving the neutron irradiation of uranium below these threshold limits are not classified as fueled experiments. Experiments involving the neutron irradiation of uranium in excess of these threshold limits, the neutron irradiation of any amount of another fissionable material, or a planned vented release of fission gases or halogens are defined as fueled experiments.

TS 3.8 provides limiting conditions for operation for fueled experiments and is revised to establish upper limits for the allowable fission rate and total number of fissions. These upper limits are based on limiting potential radiation dose from a release to three percent (3%) of the annual radiation dose limits given in 10 CFR Part 20 (see Table I below).

Table I (Ref. Table 15-1): Radiation Doses for Fueled Experiments for planned vented and accidental releases, as compared to 10 CFR Part 20 Limits

Release	Nuclide	Reactor Bldg. TEDE (rem)	Reactor Bldg.Thyroid TODE (rem)	Public TEDE (rem)
Vented	U-235	0.004		0.003
Vented	Pu-239	0.0038		0.002
Accidental	U-235	0.02	0.62	0.00027
Accidental	Pu-239	0.025	0.75	0.0003
Dose Limits for Fueled Experiments:		**0.15**	**1.5**	**0.003**
10 CFR Part 20 Limits:		***5.0*** (1)	***50*** (2)	***0.1*** (3)

(1): Total Effective Dose-Equivalent (TEDE) Annual Limit for occupationally exposed radiation worker.

(2): Total Organ Dose-Equivalent (TODE) Annual Limit to the thyroid for occupationally exposed radiation worker.

(3): TEDE Annual Limit to members of the public.

Attachment 1 to this amendment request provides the bases, assumptions, data and supporting calculations required to justify the requested limits for fueled experiments. The bounding criteria are developed in Sections 1 through 9, providing detailed analyses of both accidental and planned vented releases of fission gases and halogens, including radionuclides released, release pathways, and radiation dose from submersion, inhalation, and direct external exposure. The definition of a fueled experiment is given in Section 10, and Sections 11 through 15 provide supporting data, calculations, and conclusions.

For TS 1.2.9.e, threshold limits defining a fueled experiment using uranium are based on one percent (1%) of the annual public radiation dose given in 10 CFR Part 20 resulting from exposure to a continuous release of fission gases and halogens with the reactor building in normal ventilation for 24 hours.

For TS 3.8, upper limits on fueled experiments are established based on three percent (3%) of the annual radiation dose limits given in 10 CFR Part 20 as determined from the more restrictive of two release scenarios;

1) A vented experiment in which the fission gases and halogens are continuously filtered, delayed, and then directly exhausted into the ventilation system over the entire duration of the experiment.

2) An accidental release from an encapsulation failure which then releases an instantaneous puff of fission gases and halogens into the reactor building and is subsequently ventilated by the reactor building confinement and evacuation system for a period of 24 hours.

In the second scenario, the fueled experiment irradiation is assumed to end at the initiation of the accidental release due to the activation of the confinement and evacuation systems.

TS 3.8 also sets limits and conditions for fueled experiments that meet other TS requirements for experiments, the storage of fissionable materials, the facility emergency plan, and the facility security plan.

TS 3.8 requires controls to prevent accidental releases associated with a failure of the fueled experiment encapsulation. If planned vented releases are needed for a fueled experiment, TS 3.8 requires additional controls to limit the release rate and radiation dose. Projected activity releases and/or radiation doses do not exceed the limits established for a reportable event or emergency action levels. The amount of material requested and associated fission product activity does not exceed 10 CFR Part 37 Category 2 limits. Fissionable materials will continue to be stored as required by TS and the facility security plan and radiation protection program.

To meet planned experiment needs, a change to Section 2.B.(2) of the R-120 reactor license is requested to allow for possession of materials to be used in fueled experiments. Possession of Uranium-235,

Neptunium-237, and Plutonium-239 for fueled experiments is requested.

As a result of the analyses performed in support of this license amendment request, the following is concluded:

1) Given the low limits requested for fission rates and total fissions for fueled experiments, any doses arising from accidental or planned vented releases will be more than an order of magnitude below 10 CFR 20 limits.

2) No changes are required to the approved facility Emergency Plan, Security Plan, or Radiation Protection Program.

Technical Specification (TS) Changes:

Changes to TS 1.2.9.e. and 3.8 are requested regarding the definition and limiting conditions for operation for fueled experiments.

1.2 Definitions

1.2.9 Experiment:

e. Fueled Experiment: A fueled experiment is defined as an experiment involving any of the following:

- Neutron irradiation of uranium exceeding 1.9×10^{6} fissions per second or 1.6×10^{11} fissions.
- Neutron irradiation of any amount of neptunium or plutonium.
- A planned release of fission gases or halogens.

Fueled experiments exclude:

- Fissionable material not subjected to neutron fluence.
- Detectors containing fissionable material used in the operation of the reactor or used in an experiment, sealed sources, and fuel used in operation of the reactor.

Examples of excluded materials include manufactured detectors, sealed sources with registration certificates generated by the NRC and Agreement States, special form radioactive material as defined in 10 CFR Part 71, and NRC approved reactor fuel elements in cladding. Sealed sources are defined as sources encased in a capsule designed to prevent leakage or escape of the material from the intended use of the source or potential minor mishaps.

3.8 Operations with Fueled Experiments

Applicability

This specification applies to the operation of the reactor with a 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. Specification 3.2 pertaining to experiment reactivity worth shall be met.

b. Specifications 3.5 and 3.6 pertaining to operation of the radiation monitoring system and ventilation system shall be met during reactor operation or if moving or handling an irradiated fueled experiment.

c. Specification 3.7 pertaining to limitations on experiments shall be met.

d. Fissionable materials used in fueled experiments shall meet the following:

   i. Fissionable material physical form shall be solid, powder, or liquid.

   ii. Fission rate less than or equal to 9.6×10^{9} fissions per second.

   iii. Total number of fissions less than or equal to 1.8×10^{16}.

   iv. Vented fueled experiments shall meet the following:

      1) Solid, powder, and liquid materials shall be contained.
      2) Maximum exhaust flow rate of 3 liters per minute (lpm) that is connected directly to the reactor building ventilation system beam tube exhaust.
      3) Minimum decay time of 30 minutes before being exhausted by the reactor building ventilation system.
      4) Filtration of exhaust for particulates and halogens.
      5) Monitoring of exhaust for radioactivity before entering the reactor building ventilation system.

e. Specification 5.3 pertaining to criticality control for fueled experiments in storage shall be met.

f. Specifications 6.2.3 and 6.5 pertaining to the review of experiments shall be met.

Bases

Sections 11 and 13 of the Safety Analysis Report provide the bases for this specification.

The limitations given in Specification 3.8 ensure that:

a. 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.

b. Radiation doses from accidental or planned releases of airborne activity do not exceed three percent (3%) of the annual limits given in 10 CFR Part 20.

Specification 3.8.a provides reactivity control during irradiation.

Specification 3.8.b provides for radiation monitoring and ventilation system operation, including actuation of the confinement mode of operation should an accidental release occur during irradiation and handling of a fueled experiment.

Specification 3.8.c and 3.8.d.i provide for experimental controls to prevent release of fissionable materials and fission products.

Specification 3.8.d.ii limits radiation dose from the release of fission products to a Total Effective Dose-Equivalent (TEDE) of 0.003 rem in public areas outside the reactor building, a TEDE of 0.15 rem inside the reactor building, and a Total Organ Dose-Equivalent to the thyroid (TODE) of 1.5 rem inside the reactor building.

Specification 3.8.d.iii limits the production of long-lived fission products for safety and security concerns to levels below those given for Category 2 Quantities of Concern in 10 CFR Part 37.

Specification 3.8.d.iv provides controls for planned releases from vented experiments needed to ensure that radiation dose does not exceed three percent of the annual radiation dose limits given in 10 CFR Part 20.

Specification 3.8.e ensures that fueled experiments are stored in sub-critical configurations.

Specification 3.8.f ensures that fueled experiments are reviewed, approved, and documented as required by Specifications 6.2.3 and 6.5.

License Change:

A change to Section 2.B.(2) of the R-120 reactor license is requested to allow for possession of materials to be used in fueled experiments. Possession limits of 32 grams of Uranium-235, 1 gram of Neptunium-237, and 5 grams of Plutonium-239 for fueled experiments is requested.

Radionuclides initially present and those produced by activation of Uranium with subsequent decay include:

- Uranium: U-234, U-235, U-236, U-237, U-238, U-239
- Neptunium: Np-237, Np-238, Np-239
- Plutonium: Pu-238, Pu-239, Pu-240

U-234, U-235 and U-238 are present in natural abundances or uranium enriched in U-235.

Np-237 and Pu-239 are long-lived radionuclides.

ATTACHMENT 1

FUELED EXPERIMENT ANALYSIS

27 August 2018

INTRODUCTION

Information is provided in this analysis in support of an amendment to the Technical Specifications (TS) regarding TS 1.2.9e and TS 3.8 for fueled experiments.

Based on experiment needs, the amounts and types of fissionable materials and a maximum fission rate were determined. The maximum fission rate was used in TS 3.8 to define the upper limit for fueled experiment. Radiation doses from released fission gases and halogens are then calculated for occupied locations inside and outside the reactor building. Potential radiation dose is limited to three percent (3%) of the annual radiation dose limits given in 10 CFR Part 20; specifically, Total Effective Dose-Equivalent (TEDE) inside the reactor building is limited to 0.15 rem; Total Organ Dose-Equivalent (TODE) to the thyroid inside the reactor building is limited to 1.5 rem, and the TEDE in public areas outside the reactor building is limited to 0.003 rem. Calculations for the maximum fission rate were performed for accidental and planned releases of fission gases and halogens from the irradiation of U-235 and Pu-239. U-235 and Pu-239 were the limiting fissionable materials of those being requested for use in fueled experiments due to higher fission cross-sections and mass.

Threshold limits on fission rate and total number of fissions were established for experiments containing uranium which are based on limiting the radiation dose from a potential release to one percent (1%) of the annual public dose limit given in 10 CFR Part 20, i.e. a TEDE of 0.001 rem. The TEDE of 0.001 rem was calculated from exposure to a continuous release of fission gases and halogens with the reactor building in normal ventilation for 24 hours. The resulting threshold limits are used in TS 1.2.9e to define a fueled experiment using uranium. Experiments involving the neutron irradiation of uranium below these threshold limits are not classified as fueled experiments. Experiments involving the neutron irradiation of uranium in excess of these limits, the neutron irradiation of any amount of another fissionable material, or involving a planned vented release of fission gases or halogens are defined as fueled experiments.

The calculations performed result in a radiation dose per unit fission rate. The limiting fission rate is then determined using the criteria of not exceeding three percent of the 10 CFR 20 annual radiation dose limits. The maximum duration of the experiment and the limiting fission rate were used to define the activity of longer-lived radionuclides produced. These longer-lived radionuclides are compared to Quantities of Concern given in 10 CFR Part 37.

Sections of the analysis are listed below:

1. Assumptions
2. Saturation activity
3. Exposure time
4. Filter retention
5. Released activity
6. Atmospheric dispersion
7. Time integrated exposure
8. Dose assessment
9. Experiment limits
10. Fueled experiment definition
11. Emergency Plan
12. Security, storage, and inventory
13. Possession limits
14. Calculation results
15. Conclusions
16. References

Sections 1 through 13 provide information and equations used in the calculations.
Section 14 provides the calculation results and supporting data.
Section 15 provides conclusions used to support the requested changes to TS 3.8 and TS 1.2.9 e.

SECTION 1: ASSUMPTIONS

1.1 Fueled Experiments

1.1A Conditions for a planned vented release from a fueled experiment:

1. Fission gases and halogens are released. Release of particulate, powder, liquid, and solid material is prevented by design of the experiment.

2. A continuous, controlled release from the reactor building ventilation system during the experiment irradiation time is assumed. A minimum decay time of 30 minutes and maximum release rate of 3 liters per minute (lpm) is assumed.

3. Release occurs during the entire irradiation time.

4. Assumptions 7, 8, 9 and 16 below apply to vented experiments.

1.1B Conditions for accidental release from fueled experiments:

5. Radioactive materials are encapsulated until the time of failure.

6. Single-mode nonviolent failure of the encapsulation results in the release of radioactive noble gases and halogens into the minimum reactor building free air volume.

7. Neutron fluence rate is constant over the irradiation time and over the entire mass of the fissionable material present during the experiment irradiation time. No correction to the mass is made because of activation and fission reactions during the irradiation time. No correction is made to the fluence rate because of self-absorption by the mass of fissionable material or encapsulation materials.

8. Reactor ventilation system is in the normal mode until being activated by a radiation alarm from an abnormal release, which then places the ventilation system in confinement mode.

9. Radioactive noble gases and halogens are assumed to be present at the saturation activity from irradiation at the maximum fluence rate in the reactor experimental facilities.

10. Exposure times to personnel in the reactor building is a total of six minutes based on a radiation monitor response time of four minutes and an evacuation time of two minutes from the reactor building.

11. Exposure times to the public are 2 hours and 24 hours. Evacuation of public areas occurs within 2 hours. All released activity is removed within 24 hours.

12. The release is assumed to occur instantaneously and to be well mixed within the reactor building free air volume for accidental releases.

13. 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.

14. The minimum reactor building free air volume is assumed to be 2.25×10^3 m^3 based on reported and measured data and current design features given in TS 5.2a.

15. Confinement filter removal efficiency, or retention, is 99.97 percent for particulates and 99 percent for halogens. Retention for halogens is assumed to be 90 percent.

16. Atmospheric dispersion parameter, [X/Q] is calculated using established equations, data, and parameters given in the references. The Gaussian Plume Model (GPM) was used for all releases and release periods. Fumigation conditions were assumed to last 24 hours. The GPM was modified for calm winds. Calm winds were assumed to last 24 hours.

1.2 Non-Fueled Experiments

Conditions for irradiation of uranium at or below a TEDE of 0.001 rem:

17. The release of fission gases and halogens is assumed to be accidently released to the reactor building free air volume continuously for 24 hours using normal ventilation with no filtration.

SECTION 2: SATURATION ACTIVITY [Ref 12, 25, Section 14 Calculations 1, 2, 3, 4]

A sufficiently long production period is assumed to reach saturation activity of the fission gases and halogens. These include isotopes of Kr, Xe, I, and Br. Saturation activity was calculated using experimental facility fluence rates, reported cross-sections, and reported cumulative fission yields.

Fission product inventory for the radionuclides available for release attains saturation activity with sufficient irradiation time. Saturation activity is estimated using Equation 2-1:

$$\mathbf{A(\infty) = k\sigma\varphi NY} \qquad \textbf{Eq. 2-1}$$

where,
$A(\infty)$ is the saturation activity from thermal and non-thermal fission
k is a group conversion constant to give activity
$k = (1 \times 10^{-24}\ cm^2/barn)(1\ decay/atom)(1Ci\ /3.7 \times 10^{10}\ dps) = 2.703 \times 10^{-23}$ for Ci
or $k = 2.703 \times 10^{-29}$ for μCi
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
$N = (\text{Mass in grams, M})(\ 6.022 \times 10^{23}\ atoms/mole)(1\ mole\ /\ \text{atomic mass number, A})$
Y is the cumulative fission yield for a given radionuclide

Saturation activity is directly proportional to the fluence rate and mass of fissionable material. The fission rate, or production rate, for a given radionuclide is given by the product σφNY.

Fluence rate [Ref 5, 28]

Neutron fluence rates in experimental facilities are measured by the reactor staff following standard ASTM E261 "Standard Methods for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques" using NIST traceable materials. These measurements are made periodically, as experimental facilities change, or for specific experimental needs.

Release of halogens (I and Br fission products) is greater if no water is present since water partially retains halogens. A greater release would therefore occur for an irradiation facility outside the reactor pool, which are performed using the reactor beam tubes. Release of fission gases is not affected by the presence of water. Therefore, the maximum fluence rates measured at 1 MW operation for a reactor beam tube experiment were used in this analysis to determine the saturation activity of the fission gases and halogens that are assumed to be released.

The following fluence rates were used:

- Thermal neutron fluence rate of 1×10^{12} $cm^{-2}s^{-1}$
- Non-thermal fluence rate of 3×10^{11} $cm^{-2}s^{-1}$

Decay data [Ref 14]

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).

Cumulative fission yield data [Ref 15, 17]

Cumulative fission yields were taken from 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"
- Japan Atomic Energy Agency Nuclear Data Center Tables of Nuclear Data (JENDL data).

Fission cross-section data [Ref 16, 17, 18]

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

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

References for fission cross-section data used in this analysis:

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

SECTION 3: EXPOSURE TIME [Ref 6, 7, 22, 23]

Reactor building personnel

Evacuation time measured from various locations inside the reactor building to the evacuation exit point for several individuals ranged from 10 seconds to 1 minute or less following initiation of the reactor building evacuation signal. Evacuation followed the facility emergency plan and procedures. Also, evacuation times were calculated for an average walking pace of 3 mph for the furthest distance in the reactor building to the assembly point outside the reactor building. Both the measured time and estimated walking time gave a time of 1 minute. A travel time of 2 minutes is assumed for conservatism.

Time for released activity to reach the ventilation system and be detected by the ventilation system radiation monitors is less than 2 minutes. This is based on a ventilation system flow rate of greater than 60 feet per minute and duct length of 100 feet and detector response time of 0.5 minutes. A response time of 4 minutes is assumed for conservatism.

For personnel in the reactor building, an exposure time of 6 minutes (360 s) is used based on the time needed for detector action to activate the building evacuation alarm (4 minutes) and for personnel to physically exit the reactor building (up to 2 minutes).

Public

For the public, exposure times of 2 hours and 24 hours were used for accidental releases:

- 2 hours allows sufficient time for detection and response by facility personnel to determine affected public areas that need to be evacuated.
- 24 hours is sufficient time for the entire released activity to be vented from the reactor building (more than 10 air changes). A public exposure time of 24 hours is also associated with meeting emergency action levels given in the facility emergency plan, which would not be exceeded. After 24 h the reactor building has experienced over 10 air changes leaving a negligible fraction of the initial concentration, (e^{-10}). After 24 h the accidental release is completely ventilated.

For planned vented releases, exposure times used for the public were 2, 24, 96, and 520 hours. 520 hours is the maximum irradiation time used to limit the production of long lived radionuclides.

The exposure is presumed to last during the irradiation time. Since irradiation times may be divided, full occupancy by the public during the irradiation time is assumed.

SECTION 4: FILTER RETENTION [Ref 4, 11]

Confinement filters

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

Testing is performed per TS 4.5 on the ventilation system, including filter testing in accordance with TS 4.5 e. Maintenance and surveillance procedures are 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 retention of 0.9997 for HEPA tested with 0.3 micron aerosols and 0.99 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 of 0.99 for methyl iodine. A filter retention factor of 0.9 is used in this analysis for halogens.

Vented experiment

The exhaust from a vented experiment is filtered for halogens and particulates prior to entering the beam tube exhaust, which is connected to the reactor building ventilation system.

Filters for removal of halogens are tested by the vendor following ASTM D 3803 "Standard Test Method for Nuclear-Grade Activated Carbon" and have a reported retention factor greater than 95 percent for methyl iodine at low flow rates. A filter retention of 0.9 is used in this analysis. Particulate filters shall have a reported retention of 0.95 or higher and are used to remove particulates from the decay of noble gases and halogens. The experiment filter housing shall be sealed and, if necessary, shielded. Replacement of the experiment exhaust filters follows vendor requirements.

Air monitoring filters for vented experiments with the same retention are also used in the experiment exhaust. Air monitoring filters are in continuous use during a fueled experiment and replaced and analyzed each work week or upon observation of abnormal monitor readings, as required by facility procedures and the Radiation Protection Program.

SECTION 5: RELEASED ACTIVITY [Ref 1, 3, Section 14 Calculations 1, 2, 3, 4]

The reactor building ventilation system is operated in the normal mode for fueled experiments since these experiments may last for an extended time. Releases are monitored by the radiation monitoring system (RMS). If abnormal levels are detected by the RMS, then the reactor building ventilation system initiates an evacuation alarm and switches to confinement mode. In confinement the exhaust is filtered using a charcoal bed and particulate absorber prior to release to the environment. Operating in normal mode maintains the confinement filters as an engineered safety feature.

Reactor building free air volume [Ref 4, 11]

Measurements of the reactor building experiment area were made and give a total volume of 3.43×10^3 m^3. The free volume was measured to be 3.09×10^3 m^3 by accounting for existing equipment and experiments. The measured free volumes are greater than the FSAR value of 2.4×10^3 m^3 and TS value of 2.25×10^3 m^3. Additional equipment, modifications, or experiments in the reactor building significantly affecting free air volume are not expected.

Therefore, the value of 2.25×10^3 m^3 was used in this analysis. A lower free air volume is conservative since it increases the concentration and radiation dose.

Accidental releases

For accidental releases, the radioactive material inventory of fission gases and halogens are assumed to be completely released and then instantaneously and uniformly distributed throughout the entire reactor building free air space. The released materials are then exhausted by the ventilation system and reactor stack to the environment. The concentration inside the reactor building, decay inside the reactor building, filter retention, exhaust ventilation rate, and atmospheric dispersion are considered in the analysis.

Concentration from an accidental release

The sample is assumed to contain saturated activities of radioactive fission gases (Kr, Xe) and halogens (I, Br) at the time of encapsulation failure. The entire 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.

Initially, the release occurs for 2 minutes in normal ventilation with no filtration. Following this, the elevated concentrations of fission gases and halogens from an accidental release automatically initiate the evacuation and confinement system by the radiation monitoring system. The release is then filtered by the confinement filters for the remaining duration of the release.

The initial released concentration, C(0), in the reactor building is given by equation 5-1:

$$\mathbf{C(0) = A(\infty) / V} \qquad \textbf{Eq. 5-1}$$

where, $C(0)$ in Ci/m^3 and $A(\infty)$ in Ci
V is the minimum reactor building experiment area free air volume = 2.25×10^3 m^3

Average concentration for an accidental release inside the reactor building

Over time, the initial concentration, C(0), is removed by decay and the ventilation system. Due to the high initial concentration, the ventilation system would be in confinement mode.

Average concentration, < C >, for exposure time T is given by:

$$\mathbf{< C > = \int C(0)\, e^{-kt} dt = C(0)\, [\, (1- e^{-kT}) / (kT)]} \qquad \textbf{Eq. 5-2}$$

where, $k = \lambda + v$ in h^{-1}
$v = 0.453$ h^{-1} at 600 cfm exhaust rate in confinement
0.453 h^{-1} = (28,317ml per cubic foot)(600 ml/ min) (60 min/h) / 2.25×10^9 ml
$v = 1.41$ h^{-1} at 1870 cfm in normal ventilation
1.41 h^{-1} = (28,317 ml per cubic foot)(1870 ml/ min) (60 min/h) / 2.25×10^9 ml
t is exposure time, with limits of integration from 0 to T, in hours

Exposure times used: T is 0.066 h (4 minutes) for the exposure time in normal ventilation
T is 0.033 h (2 minutes) for the evacuation time in confinement
T is 2 h and 24 h for total public exposure time in confinement

Release rate at the reactor stack from an accidental release

The filtered release rate at the reactor stack, Q, is calculated as follows:

$$\mathbf{Q = C\, [1-R]\, F} \qquad \textbf{Eq. 5-3}$$

where, Q is the release rate in Ci/s
C is concentration in Ci/m^3, either <C> or C(0)
R is filter retention
R = 0.9 for halogens and R = 0 for noble gases, R = 0 in normal ventilation
F is the stack exhaust in m^3/s
F = 0.283 m^3/s in confinement mode and 0.883 m^3/s in normal mode

Vented experiments

A continuous, controlled release during the experiment irradiation time occurs for vented experiments.

Controls include:

- Filtration of particulates and halogens. Filters with rated retention greater than 95 percent for particulates and halogens are to be used. Retention for halogens is assumed to be 90 percent.

- Vented experiment has a minimum decay time of 0.5 hours (30 minutes). The maximum experiment exhaust rate is 5.0×10^{-5} m^3/s (3 liters per minute). At the maximum experiment exhaust rate a minimum experiment holdup volume of 0.09 m^3 (90 liters) in needed to give a 30 minute decay time. In addition, the experiment volume is designed so that activity is well mixed prior to release into the reactor building ventilation system; e.g. using a long tube or coil, a series of small air tanks, a low flow rate relative to the volume, well separated entry and exit flow ports, and baffles or diffusers within the experiment holdup volume.

- Experiment exhaust flow is routed to the ventilation system using the beam tube exhaust and controlled by dedicated equipment with local flow rate indication. The experiment exhaust is capable of being isolated. The exhaust flow tubing from the experiment to the beam tube exhaust is sealed to prevent leakage into the reactor building free air space.

- Radiation monitoring and flow rate monitoring of the experiment exhaust prior to being routed to the reactor building ventilation system is required to identify and quantify the source of the release. The release is monitored for radioactivity with indication locally and in the control room. Local alarm annunciation and alarm indication in the control room is provided.

A sudden and significant release from a vented experiment, e.g. rupture of the holdup tank, would be an accidental release as previously described. Any release of exhausted air from the vented experiment into the reactor building would be diluted by the reactor building free air volume. Radiation monitoring of the reactor building air volume and exhausted air are performed continuously by the radiation monitoring system.

Concentration from a vented experiment at the experiment exhaust

The saturation activity $A(\infty)$ is assumed to be dispersed and held in the volume of a vented experiment, w, for decay giving the decayed, unfiltered saturation concentration, $c(\infty)$:

$$\mathbf{c(\infty) = [A(\infty) / w] \exp(-\lambda t)} \qquad \textbf{Eq. 5-4}$$

where, $c(\infty)$ in Ci/m^3
$A(\infty)$ is in Ci
w is the experiment hold up volume; minimum value is 0.09 m^3 (90 liters)
Decay time, t, is 30 minutes (30 minutes = 90 liters / 3 lpm)

Concentration from a vented experiment at the reactor stack

$c(\infty)$ is diluted by the reactor building ventilation system exhaust operating in the normal mode at the reactor stack. Concentration at the reactor stack, C, is given by:

$$\mathbf{C = c(\infty)\ [p / (p+F)]} \qquad \textbf{Eq. 5-5}$$

substituting Eq. 5-4 into Eq. 5-5 gives, **C = [A (∞) / w] exp(-λt)[p / (p+F)]** **Eq. 5-6**

where, C and c(∞) in Ci/m^3
F is the normal ventilation flow rate of 0.883 m^3/s (or 1870 cfm)
p is the experiment exhaust rate and shall not exceed $5.0x10^{-5}$ m^3/s (3 lpm)
[p/(p+F)] accounts for the dilution by the normal exhaust from the reactor building

Release rate for vented experiment

The experiment exhaust is filtered and routed to the beam tube exhaust. The beam tube exhaust is connected to the reactor building ventilation system exhaust in the upper part of the reactor building.

The filtered release rate, q, from a vented experiment entering the beam tube exhaust is given by:

q = c(∞) [1-R] p **Eq. 5-7**

substituting Eq. 5-4 into Eq. 5-7gives, **q = [A (∞) / w] exp(-λt)[1-R] p; and q = Q** **Eq. 5-8**

where, q is the decayed, filtered release rate in Ci/s that enters the beam tube exhaust
p is the experiment exhaust rate in ml/s; maximum value is $5.0x10^{-5}$ m^3/s (3 lpm)
c(∞) in Ci/m^3
λ is the radioactive decay constant in 1/s
Filter retention; R = 0.9 for halogens and R = 0 for noble gases
Q is the decayed, filtered release at the reactor stack

NOTES:
- Since the flow rate is constant into and out of the experiment delay volume, w, the net loss while in the holdup volume is due to radioactive decay.
- The decayed and filtered release rate at the reactor stack, Q, is the same as the decayed and filtered experiment release rate, q, since c(∞) is diluted and exhausted by the same flow rate, F; i.e. Q = q.

Vented experiment radiation and flow monitoring

Limits on vented experiments are set for the exhaust rate, p, and experiment volume, w, to give a minimum decay time of 30 minutes. At "p" less than $5.0x10^{-5}$ m^3/s (3 lpm) or "w" more than 0.09 m^3 (90 liters), the time for decay is greater and the release rate, "q", is lower which gives a lower radiation dose.

The minimum decay time is set so that the reactor building evacuation and confinement systems are not initiated by the effluent radiation monitors. The reactor stack concentrations are decayed and diluted causing an increase in response that is detectable but below the radiation monitoring system alarm set points.

Flow rate of the experiment exhaust is controlled and has local indication. If a release above expected levels occurs, the exhaust can be stopped and isolated.

SECTION 6: ATMOSPHERIC DISPERSION [Ref 1, 12, 19, 20, 27, Section 14 Calculation 5]

Atmospheric dispersion was evaluated using reported equations given in the references for the Gaussian Plume Model (GPM) at distances from 10 m to 5000 m for all exposure times, fumigation for exposure up to 24 hours, and the GPM for a calm wind for exposure up to 24 hours.

No decay corrections are made during transport by the atmosphere following the release or decay post-production prior to release since a failure may occur anytime during the experiment.
The GPM equation was used to calculate the atmospheric dispersion parameter [X/Q] for Pasquill-Gifford (PG) weather stability classes A through F:

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z u}[e^{-\frac{y^2}{2\sigma_y^2}}][e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \quad \textbf{Eq. 6-1}$$

where, $[X/Q]_{x.y.z}$ is the atmospheric dispersion parameter for downwind location (x,y,z) in s/m^3
[X/Q] is the downwind concentration per unit release rate; 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
z and h are relative to the ground elevation of 0 m

Dispersion parameters

Pasquill-Gifford (PG) weather stability classes A through F are used for [X/Q] in the GPM 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” for downwind distances from 10 m to 5000 m. These calculated dispersion parameters for weather stability classes A through F were used in the [X/Q] equations.

Stack height

ANSI/ANS-15.7 and US NRC Regulatory Guide 1.111 were used to calculate effective stack heights. From these calculations, the effective stack height was calculated to be only slightly greater than the physical stack height. For simplicity, the actual stack height of 30 m is used in atmospheric dispersion parameter [X/Q] calculations.

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 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, [X/Q] equation 6-1 becomes:

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \quad \textbf{Eq. 6-2}$$

The plume centerline equation above accounts for a receptor location at any elevation (z) relative to the ground level.

Release time of 2 hours or longer

Sector averaging applies if the wind direction deviates sufficiently across the sector width over time, i.e. a meandering plume over the lateral "y" dimension. Sector averaging is considered valid at downwind distances (x) if $\pi x/n > 2\sigma_y$ and for periods greater than 2 hours.

On inspection, for the reactor facility stack height where the relationship $\pi x/n > 2\sigma_y$ is valid, the minimum distances are applicable for sector averaging for PG weather stability classes A through F as given in Table 6-1.

Table 6-1: Stability Classes

Stability Class	Minimum Distance (m)
A	>50,000
B	25,000
C	2,500
D, E, F	100

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

$$\overline{[X/Q]_{x,y,z}} = \sqrt{2/\pi}\,\frac{n}{2\pi x}\,\frac{f}{2\sigma_z u}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \textbf{Eq. 6-3}$$

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, f is set at 1 and μ is 1 m/s.

If sector averaging is not valid, the [X/Q] equation for the GPM was used for all elevations (z):

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \textbf{Eq. 6-4}$$

If valid, the sector average $\overline{[X/Q]}$ equation was used for all elevations (z). Re-writing with the noted assumptions for f and μ gives the following:

$$\overline{[X/Q]_{x,y,z}} = \frac{2.032}{2\sigma_z x}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \textbf{Eq. 6-5}$$

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

The following simplifications to [X/Q] GPM calculations are made regarding releases from 2 to 24 hours:

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

Fumigation

In fumigation conditions, the vertical dispersion is uniform from ground level to the stack height. Fumigation was assumed to exist for periods up to 24 h. [X/Q] for fumigation conditions for the plume centerline (y = 0 m) were calculated at a wind speed of 1 m/s for periods up to 24 h using the following equation:

$$[X/Q] = \frac{1}{\sqrt{2\pi}h\mu\sigma_y} \quad \textbf{Eq. 6-6}$$

where, h is the physical stack height of 30 m and replaces σ_z

Calm winds

Calm winds were assumed to exist for periods up to 24 hours. Calm winds have reported wind speeds less than 0.5 m/s. In calm winds the straight-line Gaussian plume model is not applicable and becomes undefined if the wind speed becomes zero. NUREG 1887 gives a model for calm winds that uses horizontal and vertical turbulence velocities (m/s) rather than normal dispersion parameters. For calm winds default turbulence velocities (σ) of 0.13 m/s are used for the wind, cross wind, and vertical turbulence.

$$[X/Q] = \frac{1}{(2\pi^{3/2})(x^2+h^2)\sigma} \quad \textbf{Eq. 6-7}$$

Release times greater than 24 hours

Release times greater than 24 hours are associated with vented experiments. Adjustments to the [X/Q] calculations were made as given in ANSI/ANS-15.7 for times from 1 to 4 days and greater than 4 days for the PG stability class frequency (S), wind direction (f), and wind speed (u).

The product [S f /u] is multiplied to the [X/Q] and $\overline{[X/Q]}$ equations given above evaluated at a wind speed of 1 m/s and then summed for all PG stability classes to give the adjusted [X/Q] value. A summary of the [X/Q] equations and adjustments are given in Table 6-2 below. Refer to Section 14 Calculation 5 in this analysis for the [X/Q] values that were calculated.

Table 6-2: Summary of the [X/Q] Equations and Adjustments

Duration	PG Stability Class	PG Stability Frequency, S	Wind Direction, f	Wind speed (m/s), u	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
24 h	Fumigation		1	1	0, centerline
24 h	Calm wind		1	0.5	0, centerline
1 to 4 days	D	0.4	1	3	Sector Averaged
	F	0.6	1	2	Sector Averaged
> 4 days	C	0.333	0.15	3	0, centerline
	D	0.333	0.15	2	Sector Averaged
	F	0.333	0.15	2	Sector Averaged

The following maximum [X/Q] values were used in this analysis to calculate time integrated exposures in occupied public areas at and beyond the site boundary:

- $8.54x10^{-3}$ s/m^3 for a release times of 2 hours and 24 hours
- $7.79x10^{-4}$ s/m^3 for a release time from 24 to 96 hours
- $9.15x10^{-5}$ s/m^3 for a release time from 96 to 520 hours

In addition, maximum [X/Q] values for specific locations of interest were used to calculate the time integrated exposure.

SECTION 7: TIME INTEGRATED EXPOSURE [Ref Section 14 Calculations 1, 2, 3, 4]

7.1 Fueled Experiments

Time integrated exposures inside the reactor building from accidental release

Time integrated exposures and Dose Conversion Factors (DCF) are used to calculate radiation dose. Time integrated exposures are given by the product of the average concentration over the exposure time and the exposure time.

Accidental releases initially occur with the reactor building in normal ventilation and then after 2 minutes the RMS or Reactor Operator activate the evacuation alarm and confinement ventilation.

The time-integrated exposure with removal by radioactive decay and ventilation system inside the reactor building from an accidental release was calculated as follows for exposure time, T:

$$\Psi_r = \langle C \rangle T \qquad \textbf{Eq. 7-1}$$

where, Ψ_r is the time integrated exposure in μCi-h/ml
<C> in μCi/ml or Ci/m^3 ; conversions are 1 μCi/ml = 1 μCi/cm^3 = 1 Ci/m^3
T is 0.066 hours inside the reactor building in normal ventilation
T is the evacuation time of 0.033 hours inside the reactor building in confinement

Calculated Ψ_r for each ventilation mode and exposure time are then summed for the total Ψ_r.

Time integrated exposures outside the reactor building from an accidental release

Accidental releases initially occur with the reactor building in normal ventilation and switches to confinement after 0.066 hours (4 minutes).

Time-integrated exposure in public areas is 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 for each exposure time, T:

$$\Psi_p = \langle C \rangle [1-R][X/Q]FT \qquad \textbf{Eq. 7-2}$$

Alternately,

$$\Psi_p = Q\ [X/Q]\ T \qquad \textbf{Eq. 7-3}$$

where, Ψ_p is the time integrated exposure in μCi-h/ml for members of the public in μCi-h/ml
<C> in μCi/ml or Ci/m^3
R = 0.9 for halogens and R = 0 for noble gases, R = 0 in normal ventilation
F is the volumetric stack exhaust rate of 0.883 m^3/s in normal ventilation and 0.283 m^3/s in confinement
[X/Q] is the atmospheric dispersion parameter in s/m^3
T is 0.066 h in normal ventilation
T is 2 hours or 24 hours in confinement
Q is Ci/s
Conversions are 1 μCi/ml = 1 μCi/cm^3 = 1 Ci/m^3

Calculated Ψ_p for each ventilation mode and exposure time are then summed for the total Ψ_p.

Time integrated exposures outside the reactor building from vented experiments

For vented experiments, the release activity is constant and continuous over the exposure time. The release is routed directly to the ventilation system, thereby not exposing occupants inside the reactor building to airborne activity. Time-integrated exposure to members of the public is given by the following equation:

$$\Psi_p = q\ [X/Q]\ T = Q\ [X/Q]\ T \qquad \text{Eq. 7-4}$$

where, Ψ_p is the time integrated exposure in μCi-h/ml for members of the public in μCi-h/ml
Conversion constants: 1E-6 Ci/μCi and1 Ci / m^3 = 1 μCi / ml
q and Q are the filtered release rate in Ci/s
[X/Q] is atmospheric dispersion parameter in s/m^3
T is 2h, 24 h, 96 h, or 520 h for public exposure time outside the reactor building

7.2 Non-Fueled Experiment – Irradiation of Uranium

Time integrated exposures for experiments with uranium

For experiments utilizing small amounts of uranium below the limits for a fueled experiment, an accidental and continuous release is assumed to occur over 24 hours with the activity dispersed into the reactor building in normal ventilation and no filtration. This case is similar to the accidental release except that the release is continuously made into the reactor building volume in normal ventilation with no filtration for 24 hours. Airborne activity monitors would indicate abnormally high readings within 24 hours.

The time integrated exposure for an experiment using uranium is given by:

$$\Psi_p = [A(\infty)/V][X/Q]FT \qquad \text{Eq. 7-5}$$

SECTION 8: DOSE ASSESSMENT

Radiation monitoring system and air sampling

Monitoring and air sampling of the reactor building exhaust and room air are continuously performed for radioactive particulates and gases as required by the reactor license and facility procedures and Radiation Protection Program. Air monitors provide indication in the control room and alarm at elevated levels. If the reactor building ventilation radiation monitors alarm, the evacuation alarm and confinement system initiate.

Setpoints are at low levels to allow for mitigation of any release and allow time for other actions to prevent activation of the emergency plan.

External dose

For radiological control purposes, external dose rates are limited and controlled by the facility radiation protection program and facility procedures consistent with experimental limitations and conditions given in TS and 10 CFR Part 20 requirements, including ALARA (As Low As Reasonably Achievable) practices. Appropriate access controls and radiation monitoring are used as required by the radiation protection program. The reactor radiation monitoring system and other radiation monitors as specified for the experiment are used to alert experimenters and reactor staff of abnormal radiation levels.

Radiation dose calculations and dose limits

Radiation doses calculated include:

- Total Effective Dose-Equivalent (TEDE) for occupants inside and outside the reactor building.
- Total Organ Dose-Equivalent (TODE) to the thyroid for occupants inside the reactor building.

Dose from accidental release is limited as follows:

- TEDE to occupants inside the reactor building is limited to 0.15 rem.
- Thyroid TODE to occupants inside the reactor building is limited to 1.5 rem.
- TEDE for members of the public is limited to 0.003 rem.

Dose from a vented experiment is limited to a TEDE of 0.003 rem for occupational workers and members of the public.

Dose from experiments using uranium with a TEDE greater than 0.001 rem to members of the public or to personnel inside the reactor building are defined as fueled experiments.

External dose calculations [Ref 4, 33, Section 14 Calculations 7, 8]

External dose from exposure to the reactor building, overhead plume, reactor stack, and ventilation system ducts were calculated using average concentrations and exposure times.

Microshield was used to determine dose outside the reactor building from the following sources:

- Contaminated air present in the reactor building.
- Overhead plume and Reactor stack.

The reactor building was modeled as a rectangular volume with a total air volume of 2.25×10^9 ml. Dimensions were set at 50 feet high by 40 feet deep and 40 feet wide. The reactor walls are made of reinforced ordinary concrete with a density of 2.35 g/ml and a thickness of 30 cm.

The overhead plume, reactor stack, and ventilation ducts were modeled as line sources with no shielding. Lengths and locations of interest for the line sources are as follows:

Overhead Plume:

- Horizontal line at a length of 100 meters at a height of 30 m.
- The highest dose point is at the line midpoint, i.e. x = 50 m and y = 0 m, at an elevation (z) of 12 m.

Reactor Stack:

- Vertical line at a length of 20 m. The exhaust duct enters the stack at a height of 10 m.
- The stack is 30 m high.
- Dose points are at the base of the entry point, i.e. z = 10 m at distances (x) from 5 m to 50 m.

Source terms for accidents are the initial concentration and average concentrations over 2 h and 24 h derived from the saturation activity dispersed within the reactor building volume in the confinement ventilation mode.

The highest dose point is opposite the midpoint of the line source, except for the stack. For the stack, occupied areas near the stack are at the bottom (or end) of the line. No correction for decay is made.

Based on the dimensions of the stack, the following relationship for activity per unit length was used for the stack line source:

Stack Activity, A(stack) = 3.93 C **Eq. 8-1**

where, Stack volume = 3.93 m^3 for 20 m length and 0.5 m diameter
C is the stack concentration in Ci/m^3
At C = 1 $Ci/m^{3,}$ A(stack) = 3.93 Ci = (3.93 m^3)(1 Ci /m^3)

Under calm winds, the activity per unit length is at a maximum. The following relationship was used for the overhead line source:

Overhead Plume Activity, A(plume) = 56.6 C **Eq. 8-2**

where, C is stack concentration in Ci / m^3
At C = 1 $Ci/m^{3,}$ A(plume) = 56.6 Ci = (1Ci/m^3)(0.283 m^3/s) (100 m / 0.5 m/s)
t, time in plume = 100 m / (0.5 m/s) = 200 s

Ventilation ducts:

Activity in the beam tube ventilation ducts, A(d), is estimated from the decayed and filtered release rate, q, and time in the ventilation system.

Time in the ventilation system is estimated to be approximately 4 s based on linear distance of 150 feet of duct and the measured linear velocity of 40 feet per second. A(d) is given by:

A(d) = 4 q **Eq. 8-3**

where, q is the decayed and filtered release rate

A(d) is distributed over multiple horizontal and vertical ducts. Maximum length of exhaust duct is 15 m. No correction for decay in the ventilation duct is made. Maximum dose rates from the ventilation exhaust ducts were calculated at the center of a line source using Microshield.

Dose from released activity [Ref Section 14 Calculations 1, 2, 3, 4, 5]

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

- Deep dose-equivalent (DDE) from submersion.
- TEDE from inhalation and submersion given by the sum of the DDE from submersion and the committed effective dose-equivalent (CEDE) from inhalation.
- Thyroid TODE is given by the sum of the DDE from submersion and committed dose-equivalent (CDE) from inhalation.

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

$$\mathbf{D = \Psi \cdot DCF \cdot \mathit{f}} \qquad \textbf{Eq. 8-4}$$

where, D is dose, in rem
Ψ is the Time Integrated Exposure (μCi-h/ml), either Ψ_r or Ψ_p
Ψ_r is taken from Eq. 7-1 and Ψ_p is taken from Eq. 7-3, Eq. 7-4, or Eq. 7-5
DCF = Dose Conversion Factor in rem/h per μCi/ml
$f = 0.1$ for submersion dose correction inside the reactor building, otherwise $f = 1$

Dose conversion factors (DCF) [Ref 21, 29, 30, 31]

Dose conversion factors (DCF) were taken from the following references:

- Inhalation DCF: Federal Guidance Report 11
- Submersion DCF: Federal Guidance Report 11 for noble gases
  Federal Guidance Report 12 for halogens
  Publication EPA400 for Xe-137 and Kr-89

Inhalation DCF were converted to rem per μCi-h/ml based on the adult breathing rate of 2.4×10^9 ml per 2000 h reported in 10 CFR Part 20 Appendix B.

Submersion dose correction [Ref 8, 9, 10, 12, 26]

Reduction of submersion dose from photons emitted by released activity inside the reactor building is made based on room dimensions using the following:

$$\boldsymbol{f = f'Gk = \mu_{en} RGk} \qquad \textbf{Eq. 8-5}$$

$$f = (4.92\times10^{-5} / \text{cm})(905 \text{ cm})(2)(1.1) = 9.8\times10^{-2} \sim 0.1$$

$$\text{Alternately,} \quad f = 2k[1-\exp(-\mu_{en}\, r)] = 2(1.1)[1- \exp(-4.92\times10^{-5} * 905)] \sim 0.1$$

where, *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* .

μ_{en} = energy absorption coefficient in air for photons, for photons with an energy of 50 keV or more this value is < $4.92\text{x}10^{-5}$ per cm.

r = effective radius of 905 cm based on the reactor building volume of $3.0\text{x}10^{9}$ ml.

G = geometry correction factor of 2 for a sphere (4π geometry) vs. hemisphere (2π geometry, semi-infinite cloud).

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.

SECTION 9: EXPERIMENT LIMITS [Ref Section 14 Calculations 1, 2, 3, 4]

Fission rate

The fission rate limit is used to control the radioactive material inventory that may be accidentally released or planned on being released during a vented experiment. The fission rate limit meets dose criteria for an accidental release or vented experiment.

For a given release, radiation dose and fission rate are calculated. To determine the fission rate, the radiation dose criterion to be met is compared to the radiation dose per unit fission rate.

Fission rate limits were calculated for U-235 and Pu-239. The lower fission rate limit is used for all fueled experiments in TS 3.8. Dose from released activity for individual fissionable materials would be the same or less than those at the fission rate limit. This allows any mixture of fissionable material to be used and met the TS dose criteria.

The fission rate limit is given by:

$$[\text{f/s}]^{\text{Limit}} = \text{Dose criterion / Calculated dose per unit fission rate} \qquad \text{Eq. 9-1}$$

where, $[\text{f/s}]^{\text{Limit}}$ is the fission rate limit in fissions per second

The dose criteria based on three percent of the annual limits given in 10 CFR Part 20 are:

- 0.15 rem TEDE to occupants inside the reactor building.
- 1.5 rem TODE to the thyroid to occupants inside the reactor building.
- 0.003 rem TEDE to members in public areas outside the reactor building.

Dose criteria for vented experiments are 0.003 rem TEDE for occupants inside the reactor building or to members of the public outside the reactor building.

Dose from experiments for experiments using uranium with a TEDE greater than 0.001 rem to members of the public or to personnel inside the reactor building are defined as fueled experiments.

The applicable dose (TEDE or Thyroid TODE) per unit f/s is used to calculate the fission rate limit.

Variance with fluence rate and time

The mass of fissionable materials used in a fueled experiment is related to the number of atoms (N). N varies inversely with φ (fluence rate) to maintain the same limiting fission rate. From the limiting fission rate, the number of atoms is calculated for the fluence rate used in the fueled experiment:

$$\mathbf{N = [f/s]^{Limit} / [\sigma\varphi]} \qquad \textbf{Eq. 9-2}$$

where, N is converted to mass of the fissionable material

The fission rate limit is based on the highest dose for a given exposure time. At other exposure times, the dose is lower. This is due to the assumption that saturation activities are always present and noting that the time integrated exposure is lower at other exposure times.

Therefore, no adjustment for irradiation time or exposure time is needed since it was accounted for in the limiting fission rate and total number of fissions.

Total number of fissions

A limit on the total number of fissions is used to prevent accumulation of higher amounts of longer lived radionuclides, including those listed in Category 2 Quantities of Concern in 10 CFR Part 37.

For a given fissionable material used in a fueled experiment, a sample may be irradiated at different fluence rates at different times provided that the total number of fissions is not exceeded.

The total number of fissions is the sum of the product of the fission rate and irradiation time T:

$$\mathbf{[fission]^{Limit} = \Sigma\ \sigma\varphi NT} \qquad \textbf{Eq. 9-3}$$

The maximum exposure time is assumed to be the same as the maximum irradiation time of 520 h at the maximum fluence rate. 520 h was used in calculating the total number fissions allowed.

Experiment controls

TS 3.8 limitations and conditions include those given previously and other TS requirements for experiments, reactivity, storage of fissionable materials and experiment reviews.

SECTION 10: FUELED EXPERIMENT DEFINITION [Ref Section 14 Calculation 9]

A fueled experiment is defined as an experiment involving any of the following:

- Neutron irradiation of uranium exceeding 1.9 x 10^6 fissions per second or 1.6 x 10^{11} fissions.
- Neutron irradiation of any amount of neptunium or plutonium.
- A planned release of fission gases or halogens.

The definition of a fueled experiment is revised to allow irradiation of materials containing small amounts of uranium. Threshold limits on fission rate and total number of fissions are established for experiments containing uranium which are based on limiting the radiation dose from a potential release to one percent (1%) of the annual public dose limit given in 10 CFR Part 20, i.e. a TEDE of 0.001 rem. Experiments involving the neutron irradiation of uranium below these threshold limits are not classified as fueled experiments. Experiments involving the neutron irradiation of uranium in excess of these threshold limits, the neutron irradiation of any amount of another fissionable material, or involving a planned vented release of fission gases or halogens are defined as fueled experiments.

For experiments with samples containing uranium:

- The release of fission gases and halogens is assumed to be accidentally released to the reactor building free air volume continuously over 24 hours during the irradiation using normal ventilation with no filtration. Airborne activity monitors would indicate abnormally high readings within 24 hours.
- The fission rate limit used to define fueled experiments is based on a TEDE of 0.001 rem to members of the public or personnel inside the reactor building.
- The total number of fissions is based on an experiment irradiation of 520 hours. The accidental 24 hour release may occur at any time during the 520 hours.

Fueled experiments exclude fissionable material not subjected to neutron fluence, detectors containing fissionable material used in the operation of the reactor or used in an experiment, sealed sources, and fuel used in operation of the reactor.

- Source Material means: (1) uranium or thorium, or any combination thereof, in any physical or chemical form or (2) ores which contain by weight one-twentieth of one percent (0.05%) or more of: (i) uranium, (ii) thorium or (iii) any combination thereof. Source material does not include special nuclear material.
- Special nuclear material means: (1) Plutonium, Uranium 233, Uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of section 51 of the Act, determines to be special nuclear material; or (2) any material artificially enriched.
- Sealed sources are defined as sources encased in a capsule designed to prevent leakage or escape of the material from the intended use of the source or potential minor mishaps. Manufactured detectors, sealed sources with registration certificates generated by the NRC and Agreement States, special form radioactive material as defined in 10 CFR Part 71, and NRC approved reactor fuel elements in cladding are examples of such excluded materials.

SECTION 11: EMERGENCY PLAN [Ref 4, 6, 7] – NO REVISION NEEDED

TODE to the thyroid and TEDE radiation dose criteria for fueled experiments are below emergency action levels (EAL) given in the facility emergency plan. These radiation doses are also below those given for the fuel handling accident release scenario assumed in Section 13 of the FSAR.

No revision to the emergency plan is needed.

SECTION 12: SECURITY, STORAGE, and INVENTORY [Ref 11, 24, 32, Section 14 Calculations 6] – NO REVISION NEEDED

The possession limits are within 10 CFR Part 37 Category 2 limits for the fissionable materials requested and associated fission product inventory. Sr-90, Cs-137, and Pm-147 activities were calculated at the fission rate limit and maximum fluence rate and irradiation time. Other fission products listed in 10 CFR Part 37 are produced in insignificant quantities due to low cumulative fission yields. At the limit for the total number of fissions, the fraction of 10 CFR Part 37 Category 2 activity limits is approximately 6.0×10^{-7}, which is well below the limiting value of 1.

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.

Fissionable materials used in fueled experiments are inventoried and accounted for as required by 10 CFR Part 70, the university broad scope license, and facility procedures.

With the limitations proposed, no revision of the security plan is needed.

SECTION 13: POSSESSION LIMITS

Possession of up to 32 grams of Uranium-235, up to 1 g of Neptunium-237, and up to 5 g of Plutonium-239 for fueled experiments is requested based on experiment needs. Experiment needs include evaluation of fissionable materials used for reactor fuel and neutron detection.

Radionuclides initially present and those produced by activation of Uranium with subsequent decay include:

- Uranium: U-234, U-235, U-236, U-237, U-238, U-239
- Neptunium: Np-237, Np-238, Np-239
- Plutonium: Pu-238, Pu-239, Pu-240

U-234, U-235 and U-238 are present in natural abundances or uranium enriched in U-235.
Np-237 and Pu-239 are long-lived radionuclides.

SECTION 14: CALCULATION RESULTS

Calculations 1 through 8 were performed for accidental and planned vented releases of fission gases and halogens from the irradiation of U-235 and Pu-239. U-235 and Pu-239 were determined to be the limiting fissionable materials of those being requested for use in fueled experiments due to higher fission cross-sections and mass.

From the calculations performed for inhalation and submersion doses, the limiting case for all fueled experiments was a planned release from a vented experiment using U-235 under the assumed conditions with a fission rate of 9.6×10^9 f/s. This maximum fission rate is used in TS 3.8.

For accidental releases, 24 hours gave the higher time integrated exposures and radiation doses. This is due to all of the activity released being ventilated from the reactor building within 24 hours. For vented experiments, $A(\infty)$ is assumed to always be present and dose is directly proportional to the product [X/Q] T. Conditions for 24 hours have the highest product of [X/Q] T and therefore the highest dose.

Calculation 9 was performed to define fueled experiments for experiments containing uranium. The fission rate calculated is used in the revised TS 1.2.9 e.

CALCULATIONS 1 and 2 – Vented Experiments using U-235 and Pu-239

Tables 14-1 and 14-2 below give the data and results for TEDE from vented experiments in public areas outside the reactor building for fueled experiments using U-235 and Pu-239, respectively.

Fission rates of 9.6×10^9 f/s for U-235 and 1.4×10^{10} f/s for Pu-239 were calculated. At the fluence rates used in this analysis, a fission rate of 9.6×10^9 f/s is equivalent to 4.95×10^{-3} g of U-235 or 3.87×10^{-3} g of Pu-239.

Table 14-1: Calculation 1 - Vented Experiment using U-235

PARAMETER VALUES

Parameter	Value	Units	Parameter	Value	Units
Nuclide	*U-235*		Target atoms, N	1.27E+19	atoms
Mass	*4.95E-03*	g	Thermal fission rate	7.42E+09	f/s
Mass Number, A	*235*	g/mol	Non-thermal fission rate	2.17E+09	f/s
Sigma thermal	*585*	b	Total fission rate	9.59E+09	f/s
Sigma non-thermal	*571*	b	Reactor volume	*2.25E+09*	ml
X/Q	*8.54E-03*	s/m^3	F confinement	*0.283*	m^3/s
Thermal flux	*1.00E+12*	cm^2/s	v confinement	*1.26E-04*	1/s
Non-thermal flux	*3.00E+11*	cm^2/s	F normal	*0.883*	m^3/s
Irradiation time	*8.64E+04*	sec	v normal	*3.92E-04*	1/s
Vented experiment exhaust	*3*	lpm	Evacuation time in confine	*120*	s
Vented experiment volume	*90*	liters	Evacuation time in normal	*240*	s
Public exposure	*24*	hours	(1-R) halogens	*0.1*	

ISOTOPIC DATA

Nuclide	Half-Life (sec)	Decay Constant (1/s)	Cumulative Yield % Thermal Fission	Cumulative Yield % Non-Thermal Fission	Eq. 2-1 Saturation Activity (μCi)
83mKr	6.70E+03	1.04E-04	*5.36E-01*	*5.75E-01*	1.41E+03
85mKr	1.61E+04	4.30E-05	*1.29E+00*	*1.36E+00*	3.38E+03
85Kr	3.39E+08	2.05E-09	*2.83E-01*	*2.96E-01*	7.41E+02
87Kr	4.57E+03	1.52E-04	*2.56E+00*	*2.54E+00*	6.63E+03
88Kr	1.02E+04	6.78E-05	*3.55E+00*	*3.43E+00*	9.14E+03
89Kr	1.89E+02	3.67E-03	*4.51E+00*	*3.97E+00*	1.14E+04
131mXe	1.03E+06	6.74E-07	*4.05E-02*	*3.54E-02*	1.02E+02
133mXe	1.89E+05	3.66E-06	*1.89E-01*	*1.97E-01*	4.95E+02
133Xe	4.53E+05	1.53E-06	*6.70E+00*	*6.71E+00*	1.74E+04
135mXe	9.18E+02	7.55E-04	*1.10E+00*	*1.26E+00*	2.95E+03
135Xe	3.28E+04	2.12E-05	*6.54E+00*	*6.58E+00*	1.70E+04
137Xe	2.29E+02	3.02E-03	*6.13E+00*	*5.98E+00*	1.58E+04
138Xe	8.46E+02	8.19E-04	*6.30E+00*	*6.00E+00*	1.62E+04
131I	6.93E+05	1.00E-06	*2.89E+00*	*3.22E+00*	7.69E+03
132I	8.26E+03	8.39E-05	*4.31E+00*	*4.66E+00*	1.14E+04
133I	7.49E+04	9.26E-06	*6.70E+00*	*6.70E+00*	1.74E+04
134I	3.16E+03	2.20E-04	*7.83E+00*	*7.63E+00*	2.02E+04
135I	2.37E+04	2.93E-05	*6.28E+00*	*6.27E+00*	1.63E+04
83Br	8.64E+03	8.02E-05	*5.40E-01*	*5.76E-01*	1.42E+03
84Br	1.91E+03	3.63E-04	*9.67E-01*	*1.01E+00*	2.53E+03

Table 14-1: Calculation 1 – continued

VENTED RELEASE DOSE SUMMARY and RESULTS				
Nuclide	Eq. 5-8, Eq. 7-4 Public Time Integrated Exposure (μCi-h/ml)	DCF (rem per μCi-h/ml)	Eq. 8-4 Public TEDE (rem)	
83mKr	1.34E-07	1.52E-02	2.03E-09	
85mKr	3.57E-07	1.10E+02	3.93E-05	
85Kr	8.44E-08	1.74E+00	1.47E-07	
87Kr	5.74E-07	5.25E+02	3.02E-04	
88Kr	9.21E-07	1.33E+03	1.23E-03	
89Kr	1.76E-09	1.20E+03	2.11E-06	
131mXe	1.16E-08	5.48E+00	6.35E-08	
133mXe	5.60E-08	1.99E+01	1.11E-06	
133Xe	1.97E-06	2.25E+01	4.43E-05	
135mXe	8.62E-08	2.79E+02	2.40E-05	
135Xe	1.86E-06	1.73E+02	3.22E-04	Fission Rate [f/s]
137Xe	7.78E-09	1.10E+02	8.56E-07	9.60E+09
138Xe	4.21E-07	7.10E+02	2.99E-04	
131I	8.74E-08	2.42E+02	2.12E-05	Fissions
132I	1.11E-07	1.49E+03	1.66E-04	1.80E+16
133I	1.95E-07	3.92E+02	7.62E-05	
134I	1.55E-07	1.73E+03	2.68E-04	rem per [f/s]
135I	1.76E-07	1.06E+03	1.87E-04	3.12E-13
83Br	1.40E-08	5.09E+00	7.12E-08	
84Br	1.50E-08	1.25E+03	1.88E-05	TEDE limit (rem)
		Total Public TEDE (rem) =	**3.00E-03**	*3.00E-03*

Supporting Calculations for U-235:

The public dose from 9.6×10^{9} f/s for U-235 is 3.0×10^{-3} rem.

Saturation activity (Reference Eq. 2-1): $A(\infty) = k\sigma\varphi NY$

Kr-87: $6.63 \times 10^{3}\ \mu Ci = (4.95 \times 10^{-3} g)(6.022 \times 10^{23} / 235\ g)(2.703 \times 10^{-29})$
$\times [(585)(1 \times 10^{12})(2.56/100)+(571)(3 \times 10^{11})(2.54/100)]$ or 6.63×10^{-3} Ci

Public Time Integrated Exposure (Reference Eq. 5-8 and Eq. 7-4):

$$\Psi_p = [A(\infty) / w] \exp(-\lambda t)[1-R]\ p\ [X/Q]T$$

Kr87: $5.74 \times 10^{-7} \mu Ci\text{-}h/ml = (6.63 \times 10^{3} \mu Ci/9 \times 10^{4} ml)(50)e^{[-(1.52E-4)(9E4)/50]}(8.54 \times 10^{-3} s/m^{3})(1 \times 10^{-6} m^{3}/ml)(24\ h)$

I133: $1.95 \times 10^{-7} \mu Ci\text{-}h/ml = (1.74 \times 10^{4}/9 \times 10^{4)}(50)e^{[-(1.53E-6)(9E4)/50]}(0.1)(8.54 \times 10^{-3})(1 \times 10^{-6})(24\ )$

Public Dose for Xe-133 (Reference Eq. 8-4): $D = \Psi \cdot DCF \cdot f$

TEDE is 4.43×10^{-5} rem = $(1.97 \times 10^{-6}\ \mu Ci\text{-}h/ml)(22.5$ rem per $\mu Ci\text{-}h/ml)$

Fission rate limit from Eq. 9-1: 9.6×10^{9} f/s = 1.0×10^{-3} rem / $(3.13 \times 10^{-13}$ rem per f/s)
Fission Limit from Eq. 9-3: 1.80×10^{16} fissions = $(9.6 \times 10^{9}$ f/s$)(1.87 \times 10^{6}$ s)

Table 14-2: Calculation 2 - Vented Experiment using Pu-239

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	*Pu239*		Target atoms, N	1.43E+19	atoms
Mass	*5.67E-03*	g	Thermal fission rate	1.07E+10	f/s
Mass Number, A	*239*	g/mol	Non-thermal fission rate	3.38E+09	f/s
Sigma thermal	*748*	b	Total fission rate	1.41E+10	f/s
Sigma non-thermal	*789*	b	Reactor volume	*2.25E+09*	ml
X/Q	*8.54E-03*	s/m^3	F confinement	*0.283*	m^3/s
Thermal flux	*1.00E+12*	cm^2/s	v confinement	*1.26E-04*	1/s
Non-thermal flux	*3.00E+11*	cm^2/s	F normal	*0.883*	m^3/s
Irradiation time	*8.64E+04*	sec	v normal	*3.92E-04*	1/s
Vented experiment exhaust	*3*	lpm	Evacuation time in confine	*120*	s
Vented experiment volume	*90*	liters	Evacuation time in normal	*240*	s
Public exposure	*24*	h	(1-R) halogens	*0.1*	

ISOTOPIC DATA					
Nuclide	**Half-Life (sec)**	**Decay Constant (1/s)**	**Cumulative Yield % Thermal Fission**	**Cumulative Yield % Non-Thermal Fission**	**Eq. 2-1 Saturation Activity (μCi)**
83mKr	6.70E+03	1.04E-04	*2.97E-01*	*3.15E-01*	1.14E+03
85mKr	1.61E+04	4.30E-05	*5.63E-01*	*5.94E-01*	2.17E+03
85Kr	3.39E+08	2.05E-09	*1.23E-01*	*1.38E-01*	4.81E+02
87Kr	4.57E+03	1.52E-04	*9.89E-01*	*1.04E+00*	3.81E+03
88Kr	1.02E+04	6.78E-05	*1.27E+00*	*1.29E+00*	4.85E+03
89Kr	1.89E+02	3.67E-03	*1.45E+00*	*1.45E+00*	5.52E+03
131mXe	1.03E+06	6.74E-07	*4.24E-02*	*4.27E-02*	1.62E+02
133mXe	1.89E+05	3.66E-06	*2.31E-01*	*2.45E-01*	8.92E+02
133Xe	4.53E+05	1.53E-06	*7.02E+00*	*6.97E+00*	2.66E+04
135mXe	9.18E+02	7.55E-04	*1.84E+00*	*2.08E+00*	7.21E+03
135Xe	3.28E+04	2.12E-05	*7.60E+00*	*7.54E+00*	2.89E+04
137Xe	2.29E+02	3.02E-03	*6.01E+00*	*5.58E+00*	2.24E+04
138Xe	8.46E+02	8.19E-04	*5.17E+00*	*4.71E+00*	1.92E+04
131I	6.93E+05	1.00E-06	*3.86E+00*	*3.88E+00*	1.47E+04
132I	8.26E+03	8.39E-05	*5.39E+00*	*5.32E+00*	2.04E+04
133I	7.49E+04	9.26E-06	*6.97E+00*	*6.91E+00*	2.65E+04
134I	3.16E+03	2.20E-04	*7.41E+00*	*7.11E+00*	2.79E+04
135I	2.37E+04	2.93E-05	*6.54E+00*	*6.08E+00*	2.44E+04
83Br	8.64E+03	8.02E-05	*2.97E-01*	*3.15E-01*	1.14E+03
84Br	1.91E+03	3.63E-04	*4.29E-01*	*4.63E-01*	1.66E+03

Table 14-2: Calculation 2 – continued

VENTED RELEASE DOSE SUMMARY and RESULTS				
Nuclide	Eq. 5-8, Eq. 7-4 Public Time Integrated Exposure (μCi-h/ml)	DCF (rem per μCi-h/ml)	Eq. 8-4 Public Dose (rem)	
83mKr	1.08E-07	1.52E-02	1.64E-09	
85mKr	2.29E-07	1.10E+02	2.52E-05	
85Kr	5.48E-08	1.74E+00	9.53E-08	
87Kr	3.30E-07	5.25E+02	1.73E-04	
88Kr	4.89E-07	1.33E+03	6.51E-04	
89Kr	8.54E-10	1.20E+03	1.02E-06	
131mXe	1.84E-08	5.48E+00	1.01E-07	
133mXe	1.01E-07	1.99E+01	2.01E-06	
133Xe	3.02E-06	2.25E+01	6.79E-05	
135mXe	2.11E-07	2.79E+02	5.88E-05	
135Xe	3.16E-06	1.73E+02	5.48E-04	Fission Rate [f/s]
137Xe	1.11E-08	1.10E+02	1.22E-06	1.41E+10
138Xe	5.01E-07	7.10E+02	3.56E-04	
131I	1.67E-07	2.42E+02	4.05E-05	Fissions
132I	2.00E-07	1.49E+03	2.98E-04	2.63E+16
133I	2.96E-07	3.92E+02	1.16E-04	
134I	2.14E-07	1.73E+03	3.70E-04	rem per [f/s]
135I	2.64E-07	1.06E+03	2.81E-04	2.13E-13
83Br	1.13E-08	5.09E+00	5.74E-08	
84Br	9.84E-09	1.25E+03	1.23E-05	Dose limit, rem
		Total Public Dose (rem) =	3.00E-03	*3.00E-03*

Supporting Calculations for Pu-239:

3.87×10^{-3} g of Pu-239 gives a fission rate of 9.6×10^{9} f/s:

$3.87 \times 10^{-3}\text{g} = (5.67 \times 10^{-3}\text{g})(9.6 \times 10^{9}\text{ f/s})/(1.41 \times 10^{10}\text{ f/s})$

The public dose at 9.6×10^{9} f/s for Pu-239 is: 2.04×10^{-3} rem = $(9.6 \times 10^{9}$ f/s$)(2.13 \times 10^{-13}$ rem per f/s)

CALCULATIONS 3 and 4: Accidental Release from Experiment using Pu-239 and U-235

Tables 14-3 through 14-6 below give the data and results for radiation dose from accidental releases from experiments using Pu-239.

Tables 14-7 through 14-9 give the data and results for radiation dose from accidental releases from experiments using U-235.

For accidental releases, the thyroid TODE dose criteria for fueled experiments of 1.5 rem is limiting for both U-235 and Pu-239. Fission rates of 1.90×10^{10} f/s for Pu-239 and 2.32×10^{10} f/s for U-235.

A fission rate of 1.90×10^{10} f/s is used to compare results from Pu-239 and U-235.

Table 14-3: Calculation 3 -Accidental release from experiment using Pu-239

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	*Pu239*		Target atoms, N	1.93E+19	atoms
Mass	*7.65E-03*	g	Thermal fission rate	1.44E+10	f/s
Mass Number, A	*239*	g/mol	Non-thermal fission rate	4.56E+09	f/s
Sigma thermal	*748*	b	Total fission rate	1.90E+10	f/s
Sigma non-thermal	*789*	b	Reactor volume	*2.25E+09*	ml
X/Q =	*8.54E-03*	s/m^3	F confinement	*0.283*	m^3/s
Thermal flux	*1.00E+12*	cm^2/s	v confinement	*1.26E-04*	1/s
Non-thermal flux	*3.00E+11*	cm^2/s	F normal	*0.883*	m^3/s
Irradiation time	*8.64E+04*	sec	v normal	*3.92E-04*	1/s
Vented experiment exhaust	*3*	lpm	Evacuation time in confine	*120*	s
			Evacuation time in normal	*240*	s
Vented experiment volume	*90*	Liters	NG reactor correction	*0.1*	
Public Hours	*24*	h	(1-R) halogens	*0.1*	

ISOTOPIC DATA

Nuclide	**Half-Life (sec)**	**Decay Constant (1/s)**	**Cumulative Yield % Thermal Fission**	**Cumulative Yield % Non-Thermal Fission**	**Eq. 2-1 Saturation Activity (μCi)**
83mKr	6.70E+03	1.04E-04	*2.97E-01*	*3.15E-01*	1.54E+03
85mKr	1.61E+04	4.30E-05	*5.63E-01*	*5.94E-01*	2.93E+03
85Kr	3.39E+08	2.05E-09	*1.23E-01*	*1.38E-01*	6.49E+02
87Kr	4.57E+03	1.52E-04	*9.89E-01*	*1.04E+00*	5.13E+03
88Kr	1.02E+04	6.78E-05	*1.27E+00*	*1.29E+00*	6.54E+03
89Kr	1.89E+02	3.67E-03	*1.45E+00*	*1.45E+00*	7.45E+03
131mXe	1.03E+06	6.74E-07	*4.24E-02*	*4.27E-02*	2.18E+02
133mXe	1.89E+05	3.66E-06	*2.31E-01*	*2.45E-01*	1.20E+03
133Xe	4.53E+05	1.53E-06	*7.02E+00*	*6.97E+00*	3.59E+04
135mXe	9.18E+02	7.55E-04	*1.84E+00*	*2.08E+00*	9.73E+03
135Xe	3.28E+04	2.12E-05	*7.60E+00*	*7.54E+00*	3.89E+04
137Xe	2.29E+02	3.02E-03	*6.01E+00*	*5.58E+00*	3.03E+04
138Xe	8.46E+02	8.19E-04	*5.17E+00*	*4.71E+00*	2.59E+04
131I	6.93E+05	1.00E-06	*3.86E+00*	*3.88E+00*	1.98E+04
132I	8.26E+03	8.39E-05	*5.39E+00*	*5.32E+00*	2.76E+04
133I	7.49E+04	9.26E-06	*6.97E+00*	*6.91E+00*	3.57E+04
134I	3.16E+03	2.20E-04	*7.41E+00*	*7.11E+00*	3.76E+04
135I	2.37E+04	2.93E-05	*6.54E+00*	*6.08E+00*	3.30E+04
83Br	8.64E+03	8.02E-05	*2.97E-01*	*3.15E-01*	1.54E+03
84Br	1.91E+03	3.63E-04	*4.29E-01*	*4.63E-01*	2.24E+03

Table 14-4: Calculation 3 - Time integrated exposures for Pu-239 for a public exposure time of 24 h

Nuclide	Eq. 5-2, Eq. 7-1 Time Integrated Exposure - Confinement Reactor (µCi-h/ml)	Eq. 5-2, Eq. 7-1 Time Integrated Exposure - Confinement Public (µCi-h/ml)	Eq. 5-2, Eq. 7-2 Time Integrated Exposure- Normal Reactor (µCi-h/ml)	Eq. 5-2, Eq. 7-2 Time Integrated Exposure-Normal Public (µCi-h/ml)
83mKr	2.26E-08	2.01E-09	4.31E-08	3.25E-10
85mKr	4.29E-08	5.17E-09	8.23E-08	6.21E-10
85Kr	9.54E-09	1.54E-09	1.84E-08	1.38E-10
87Kr	7.48E-08	5.52E-09	1.43E-07	1.07E-09
88Kr	9.58E-08	1.01E-08	1.84E-07	1.38E-09
89Kr	8.86E-08	5.86E-10	1.41E-07	1.06E-09
131mXe	3.21E-09	5.15E-10	6.17E-09	4.65E-11
133mXe	1.77E-08	2.77E-09	3.40E-08	2.56E-10
133Xe	5.28E-07	8.42E-08	1.02E-06	7.66E-09
135mXe	1.37E-07	3.30E-09	2.52E-07	1.90E-09
135Xe	5.72E-07	7.91E-08	1.10E-06	8.28E-09
137Xe	3.74E-07	2.87E-09	6.12E-07	4.62E-09
138Xe	3.63E-07	8.20E-09	6.67E-07	5.03E-09
131I	2.91E-07	4.66E-09	5.60E-07	4.22E-09
132I	4.03E-07	3.92E-09	7.71E-07	5.81E-09
133I	5.24E-07	7.89E-09	1.01E-06	7.60E-09
134I	5.46E-07	3.25E-09	1.04E-06	7.82E-09
135I	4.84E-07	6.34E-09	9.29E-07	7.00E-09
83Br	2.26E-08	2.24E-10	4.33E-08	3.26E-10
84Br	3.23E-08	1.37E-10	6.08E-08	4.58E-10

Supporting Calculations:

Saturation Activity, A(∞) (Reference Eq. 2-1): $A(\infty) = k\,\sigma\,\varphi\,N\,Y$

I-131: $1.98\text{x}10^{4}$ µCi = $(7.65\text{x}10^{-3}\text{ g})(6.022\text{x}10^{23}/239\text{ g})(2.703\text{x}10^{-29})$
x $[(748)(1\text{x}10^{12})(3.86/100)+(789)(3\text{x}10^{11})(3.88/100)]$, or $1.98\text{x}10^{-2}$ Ci

Time integrated exposure (Reference Eq. 5-2 and Eq. 7-1):

$$\Psi_r = \langle C\rangle T \text{ and } \langle C \rangle = C(0)[\,(1- e^{-kT}) / (kT)] \text{ or } \Psi_r = C(0)[\,(1- e^{-kT}) / k]$$

where:

Ψr for each ventilation mode and exposure time is summed for the reactor building:
$k = \lambda + v = 1.4176$ per h in normal ventilation and 0.45858 per h in confinement
T = 0.066 h in normal ventilation and 0.033 h in confinement

Xe-133: Ψr $= (3.59\text{x}10^{4}\ \mu\text{Ci} / 2.25\text{x}10^{9}\text{ ml})$
x $\{[1-\exp((-1.4176)(0.066)) / 1.4176] + [1-\exp((-0.45858)(0.033))/0.45858]\}$
$= (5.28\text{x}10^{-7} + 1.02\text{x}10^{-6})$ µCi-h/ml
$= 1.548\text{x}10^{-6}$ µCi-h/ml

Time integrated exposure (Reference Eq. 5-2 and Eq. 7-2):

Ψ_p = <C>[1-R][X/Q]FT

where:

Ψp for each ventilation mode and exposure time is summed for the public:
k = λ+ v = 1.4176 per h in normal ventilation and 0.45858 per h in confinement
T = 0.066 h in normal ventilation and 24 h in confinement
F = 0.883 m^3/s in normal ventilation and 0.283 m^3/s in confinement
[X/Q] = 8.54×10^{-3} s/m^3 for T up to 24 h
R = 0

Xe-133: Ψp = (3.59×10^4 μCi / 2.25E9 ml)(8.54×10^{-3})
x {(0.883)[1-exp((-1.4176)(0.066)) / 1.4176]+ (0.283)[1-exp((-0.45858)(24))/0.45858]}
= (8.42×10^{-8} +7.66×10^{-9}) μCi-h/ml = 9.186×10^{-8} μCi-h/ml

Dose calculations (Reference Eq. 8-4): D = Ψ · DCF · *f*

Xe-133 dose inside the reactor building:

3.47×10^{-6} rem = [(5.28×10^{-7} + 1.02×10^{-6}) μCi-h/ml](22.5 rem per μCi-h/ml)(0.1)
= 1.19×10^{-6} + 2.28×10^{-6}) rem

Xe-133 dose outside the reactor building:

2.06×10^{-6} rem = [(8.42×10^{-8} +7.66×10^{-9}) μCi-h/ml]((22.5rem per μCi-h/ml)
= (1.89×10^{-6} + 1.72×10^{-7}) rem

I-131 dose inside the reactor building:

TEDE is 3.36×10^{-2} rem = (2.91×10^{-7} + 5.60×10^{-7})(3.95×10^4+0.1(242))
= (1.15×10^{-2}+ 2.21×10^{-2}) rem

Thyroid TODE is 1.10 rem = (2.91×10^{-7} + 5.60×10^{-7})(1.3×10^6+0.1(242))
= (0.378 +0.726) rem

I-131 dose outside the reactor building:

TEDE is 3.53×10^{-4} rem = (4.66×10^{-9} + 4.22×10^{-9})(3.95×10^4 + 242)
= (1.85×10^{-4} + 1.68×10^{-4}) rem

Fission rate (Reference Eq. 9-1):

The fission rate limit for an accidental release from Pu-239 irradiation is based on the thyroid dose inside the reactor building:

$[f/s]^{Limit}$ = Dose criterion / Calculated dose per unit fission rate

1.90×10^{-10} f/s = (1.5 rem) / (7.90×10^{-11} rem per f/s)

Table 14-5: Calculation 3 – Pu-239 Dose Calculation Results

Nuclide	Effective Inhaltn. DCF (rem per μCi-h/ml)	Thyroid Inhaltn. DCF (rem per μCi-h/ml)	Submer. DCF (rem per μCi-h/ml)	Eq. 8-4 Reactor Vent. Confine. TEDE (rem)	Eq. 8-4 Reactor Vent. Confine. Thyroid Dose (rem)	Eq. 8-4 Public Vent. Confine. TEDE (rem)	Eq. 8-4 Reactor Normal Vent. TEDE (rem)	Eq. 8-4 Reactor Normal Vent. Thyroid TODE (rem)	Eq. 8-4 Public Normal Vent. TEDE (rem)
83mKr			1.52E-02	3.42E-11	3.42E-11	3.05E-11	6.55E-11	6.55E-11	4.93E-12
85mKr			1.10E+02	4.73E-07	4.73E-07	5.71E-07	9.08E-07	9.08E-07	6.84E-08
85Kr			1.74E+00	1.66E-09	1.66E-09	2.68E-09	3.19E-09	3.19E-09	2.41E-10
87Kr			5.25E+02	3.93E-06	3.93E-06	2.90E-06	7.49E-06	7.49E-06	5.65E-07
88Kr			1.33E+03	1.28E-05	1.28E-05	1.34E-05	2.45E-05	2.45E-05	1.84E-06
89Kr			1.20E+03	1.06E-05	1.06E-05	7.03E-07	1.69E-05	1.69E-05	1.28E-06
131mXe			5.48E+00	1.76E-09	1.76E-09	2.82E-09	3.38E-09	3.38E-09	2.54E-10
133mXe			1.99E+01	3.52E-08	3.52E-08	5.52E-08	6.77E-08	6.77E-08	5.10E-09
133Xe			2.25E+01	1.19E-06	1.19E-06	1.89E-06	2.28E-06	2.28E-06	1.72E-07
135mXe			2.79E+02	3.81E-06	3.81E-06	9.19E-07	7.02E-06	7.02E-06	5.29E-07
135Xe			1.73E+02	9.90E-06	9.90E-06	1.37E-05	1.90E-05	1.90E-05	1.43E-06
137Xe			1.10E+02	4.11E-06	4.11E-06	3.16E-07	6.74E-06	6.74E-06	5.08E-07
138Xe			7.10E+02	2.58E-05	2.58E-05	5.82E-06	4.74E-05	4.74E-05	3.57E-06
131I	3.95E+04	1.30E+06	2.42E+02	1.15E-02	3.78E-01	1.85E-04	2.21E-02	7.26E-01	1.68E-04
132I	4.57E+02	7.73E+03	1.49E+03	2.44E-04	3.17E-03	7.64E-06	4.68E-04	6.08E-03	1.13E-05
133I	7.02E+03	2.16E+05	3.92E+02	3.70E-03	1.13E-01	5.84E-05	7.11E-03	2.18E-01	5.63E-05
134I	1.58E+02	1.28E+03	1.73E+03	1.81E-04	7.93E-04	6.14E-06	3.43E-04	1.51E-03	1.48E-05
135I	1.47E+03	3.76E+04	1.06E+03	7.65E-04	1.82E-02	1.61E-05	1.47E-03	3.50E-02	1.78E-05
83Br	1.03E+02		5.09E+00	2.35E-06	1.15E-08	2.43E-08	4.50E-06	2.20E-08	3.54E-08
84Br	1.01E+02		1.25E+03	7.30E-06	4.04E-06	1.85E-07	1.37E-05	7.62E-06	6.20E-07
Total				***1.65E-02***	***5.13E-01***	***3.14E-04***	***3.17E-02***	***9.86E-01***	***2.78E-04***

Table 14-6: Calculation 3 – Fueled Experiment Summary for Pu-239

Parameter	TEDE rem Reactor Building	Thyroid TODE rem Reactor Building	TEDE rem Public Areas
Total dose in rem	4.81E-02	1.50E+00	5.92E-04
rem per f/s	2.54E-12	7.90E-11	3.12E-14
Dose limit in rem	0.15	1.5	3.00E-03
Fission rate limit: Eq. 9-1	***5.91E+10 f/s***	***1.90E+10 f/s***	***9.61E+10 f/s***

Analysis Notes:

At 9.6×10^{9} f/s, the doses are 0.5 times those listed in Table 14-6

Total dose in rem: TEDE for Public Areas = 5.92×10^{-4} rem = $3.14\times10^{-4} + 2.78\times10^{-4}$ rem

Rem per f/s: TEDE for Reactor Building = $2.54\times10^{-12} = 4.81\times10^{-2}$ rem / 1.90×10^{10} f/s

Table 14-7: Calculation 4 - Accident release from experiment using U-235

PARAMETER VALUES

Parameter	Value	Units	Parameter	Value	Units
Nuclide	*U-235*		Target atoms, N	2.51E+19	atoms
Mass	*9.8E-03*	g	Thermal fission rate	1.47E+10	f/s
Mass Number, A	*235*	g/mol	Non-thermal fission rate	4.30E+09	f/s
Sigma thermal	*585*	b	Total fission rate	1.90E+10	f/s
Sigma non-thermal	*571*	b	Reactor volume	*2.25E+09*	ml
X/Q =	*8.54E-03*	s/m^3	F confinement	*0.283*	m^3/s
Thermal flux	*1.00E+12*	cm^2/s	v confinement	*1.26E-04*	1/s
Non-thermal flux	*3.00E+11*	cm^2/s	F normal	*0.883*	m^3/s
Irradiation time	*8.64E+04*	sec	v normal	*3.92E-04*	1/s
Vented experiment exhaust	*3*	lpm	Evacuation time	*120*	s
Vented experiment volume	*90*	liters	NG reactor correction	*0.1*	
Public Hours	*24*	h	(1-R) halogens	*0.1*	

ISOTOPIC DATA

Nuclide	Half-Life (sec)	Decay Constant (1/s)	Cumulative Yield % Thermal Fission	Cumulative Yield % Non-Thermal Fission	Eq. 2-1 Saturation Activity (μCi)
83mKr	6.70E+03	1.04E-04	*2.97E-01*	*3.15E-01*	2.80E+03
85mKr	1.61E+04	4.30E-05	*5.63E-01*	*5.94E-01*	6.70E+03
85Kr	3.39E+08	2.05E-09	*1.23E-01*	*1.38E-01*	1.47E+03
87Kr	4.57E+03	1.52E-04	*9.89E-01*	*1.04E+00*	1.31E+04
88Kr	1.02E+04	6.78E-05	*1.27E+00*	*1.29E+00*	1.81E+04
89Kr	1.89E+02	3.67E-03	*1.45E+00*	*1.45E+00*	2.25E+04
131mXe	1.03E+06	6.74E-07	*4.24E-02*	*4.27E-02*	2.02E+02
133mXe	1.89E+05	3.66E-06	*2.31E-01*	*2.45E-01*	9.80E+02
133Xe	4.53E+05	1.53E-06	*7.02E+00*	*6.97E+00*	3.44E+04
135mXe	9.18E+02	7.55E-04	*1.84E+00*	*2.08E+00*	5.84E+03
135Xe	3.28E+04	2.12E-05	*7.60E+00*	*7.54E+00*	3.36E+04
137Xe	2.29E+02	3.02E-03	*6.01E+00*	*5.58E+00*	3.13E+04
138Xe	8.46E+02	8.19E-04	*5.17E+00*	*4.71E+00*	3.20E+04
131I	6.93E+05	1.00E-06	*3.86E+00*	*3.88E+00*	1.52E+04
132I	8.26E+03	8.39E-05	*5.39E+00*	*5.32E+00*	2.25E+04
133I	7.49E+04	9.26E-06	*6.97E+00*	*6.91E+00*	3.44E+04
134I	3.16E+03	2.20E-04	*7.41E+00*	*7.11E+00*	4.00E+04
135I	2.37E+04	2.93E-05	*6.54E+00*	*6.08E+00*	3.22E+04
83Br	8.64E+03	8.02E-05	*2.97E-01*	*3.15E-01*	2.81E+03
84Br	1.91E+03	3.63E-04	*4.29E-01*	*4.63E-01*	5.01E+03

Table 14-8: Calculation 4 - U-235 Accident release for a public exposure time of 24 h at 6.33x10^9 f/s

Nuclide	Eq. 5-2, Eq. 7-1 Time Integrated Exposure - Confinement Reactor (μCi-h/ml)	Eq. 5-2, Eq. 7-1 Time Integrated Exposure - Confinement Public (μCi-h/ml)	Eq. 5-2, Eq. 7-2 Time Integrated Exposure- Normal Reactor (μCi-h/ml)	Eq. 5-2, Eq. 7-2 Time Integrated Exposure-Normal Public (μCi-h/ml)
83mKr	4.09E-08	3.64E-09	7.81E-08	5.89E-10
85mKr	9.83E-08	1.18E-08	1.89E-07	1.42E-09
85Kr	2.16E-08	3.48E-09	4.15E-08	3.13E-10
87Kr	1.91E-07	1.41E-08	3.64E-07	2.75E-09
88Kr	2.65E-07	2.79E-08	5.07E-07	3.82E-09
89Kr	2.68E-07	1.77E-09	4.26E-07	3.21E-09
131mXe	2.97E-09	4.77E-10	5.71E-09	4.30E-11
133mXe	1.44E-08	2.26E-09	2.77E-08	2.09E-10
133Xe	5.06E-07	8.06E-08	9.73E-07	7.33E-09
135mXe	8.21E-08	1.98E-09	1.51E-07	1.14E-09
135Xe	4.94E-07	6.83E-08	9.48E-07	7.15E-09
137Xe	3.86E-07	2.97E-09	6.33E-07	4.77E-09
138Xe	4.48E-07	1.01E-08	8.22E-07	6.20E-09
131I	2.24E-07	3.58E-09	4.30E-07	3.24E-09
132I	3.30E-07	3.21E-09	6.31E-07	4.75E-09
133I	5.05E-07	7.60E-09	9.72E-07	7.32E-09
134I	5.80E-07	3.45E-09	1.10E-06	8.30E-09
135I	4.73E-07	6.20E-09	9.08E-07	6.85E-09
83Br	4.11E-08	4.07E-10	7.88E-08	5.94E-10
84Br	7.21E-08	3.06E-10	1.36E-07	1.02E-09

Nuclide	Effective Inhaltn. DCF (rem per μCi-h/ml)	Thyroid Inhaltn. DCF (rem per μCi-h/ml)	Submer. DCF (rem per μCi-h/ml)	Eq. 8-4 Reactor Confine. TEDE (rem)	Eq. 8-4 Reactor Confine. Thyroid Dose (rem)	Eq. 8-4 Public Confine. TEDE (rem)	Eq. 8-4 Reactor Normal TEDE (rem)	Eq. 8-4 Reactor Normal Thyroid TODE (rem)	Eq. 8-4 Public Normal TEDE (rem)
83mKr			1.52E-02	6.20E-11	6.20E-11	5.52E-11	1.19E-10	1.19E-10	8.93E-12
85mKr			1.10E+02	1.08E-06	1.08E-06	1.31E-06	2.08E-06	2.08E-06	1.57E-07
85Kr			1.74E+00	3.75E-09	3.75E-09	6.05E-09	7.22E-09	7.22E-09	5.44E-10
87Kr			5.25E+02	1.00E-05	1.00E-05	7.42E-06	1.91E-05	1.91E-05	1.44E-06
88Kr			1.33E+03	3.53E-05	3.53E-05	3.71E-05	6.76E-05	6.76E-05	5.09E-06
89Kr			1.20E+03	3.22E-05	3.22E-05	2.13E-06	5.12E-05	5.12E-05	3.86E-06
131mXe			5.48E+00	1.63E-09	1.63E-09	2.61E-09	3.13E-09	3.13E-09	2.36E-10
133mXe			1.99E+01	2.87E-08	2.87E-08	4.50E-08	5.51E-08	5.51E-08	4.16E-09
133Xe			2.25E+01	1.14E-06	1.14E-06	1.81E-06	2.18E-06	2.18E-06	1.65E-07
135mXe			2.79E+02	2.29E-06	2.29E-06	5.51E-07	4.21E-06	4.21E-06	3.17E-07
135Xe			1.73E+02	8.55E-06	8.55E-06	1.18E-05	1.64E-05	1.64E-05	1.24E-06
137Xe			1.10E+02	4.25E-06	4.25E-06	3.26E-07	6.96E-06	6.96E-06	5.25E-07
138Xe			7.10E+02	3.18E-05	3.18E-05	7.18E-06	5.84E-05	5.84E-05	4.40E-06
131I	3.95E+04	1.30E+06	2.42E+02	8.84E-03	2.90E-01	1.42E-04	1.70E-02	5.58E-01	1.29E-04
132I	4.57E+02	7.73E+03	1.49E+03	2.00E-04	2.60E-03	6.25E-06	3.83E-04	4.97E-03	9.27E-06
133I	7.02E+03	2.16E+05	3.92E+02	3.57E-03	1.09E-01	5.63E-05	6.85E-03	2.10E-01	5.42E-05
134I	1.58E+02	1.28E+03	1.73E+03	1.92E-04	8.42E-04	6.52E-06	3.64E-04	1.60E-03	1.57E-05
135I	1.47E+03	3.76E+04	1.06E+03	7.48E-04	1.78E-02	1.57E-05	1.44E-03	3.42E-02	1.74E-05
83Br	1.03E+02		5.09E+00	4.28E-06	2.09E-08	4.42E-08	8.19E-06	4.01E-08	6.44E-08
84Br	1.01E+02		1.25E+03	1.63E-05	9.04E-06	4.14E-07	3.07E-05	1.70E-05	1.39E-06
Total				***1.37E-02***	***4.21E-01***	***2.97E-04***	***2.63E-02***	***8.09E-01***	***2.44E-04***

Table 14-9: Calculation 4 - Fueled Experiment Summary for U-235

Parameter	Effective rem Reactor Building	Thyroid rem Reactor Building	Effective rem Public Areas
Total dose in rem	4.00E-02	1.23E+00	5.41E-04
rem per f/s	2.11E-12	6.47E-11	2.85E-14
Dose limit in rem	1.50E-01	1.5	3.00E-03
Fission rate limit (f/s): Eq. 9-1	***7.13E+10***	***2.32E+10***	***1.05E+11***

NOTE: At 9.6×10^9 f/s, the doses are 0.5 times those listed in Table 14-9

CALCULATION 5: Radiation Doses (TEDE) for Specific Public Locations of Interest

Tables 14-10 gives [X/Q] values for specific public locations of interest that were calculated as described in Section 6. Maximum [X/Q] values were 8.54×10^{-3} s/m^3 for periods up to 24 h, 7.79×10^{-4} s/m^3 for 96 h, and 9.15×10^{-5} s/m^3 for 520 h.

Tables 14-11 and 14-12 give the TEDE for public areas from U-235 and Pu-239 for the limiting fission rate of 9.6×10^9 f/s.

TEDE in publicly occupied areas outside the reactor building were calculated at a fission rate of 9.6×10^9 f/s to be 3.0×10^{-4} rem or less from an accidental release and 3.0×10^{-3} rem or less from a vented experiment.

Table 14-10: Calculation 5 - [X/Q] values for specific public locations of interest

Building or Location	Distance x (m)	Height z (m)	Eq. 6-6 [X/Q] Fumign. Up to 24 h (s/m^3)	Eq. 6-7 [X/Q] Calm Wind Up to 24 h (s/m^3)	Eq. 6-4 [X/Q] GPM 2 h (s/m^3)	Eq. 6-4 Eq. 6-5 [X/Q] GPM 24 h (s/m^3)	Eq. 6-4 Eq. 6-5 [X/Q] GPM 96 h (s/m^3)	Eq. 6-4 Eq. 6-5 [X/Q] GPM 520 h (s/m^3)
All	30 to 100	up to 12	8.54E-03	4.63E-04	2.31E-04	2.31E-04	1.39E-07	1.56E-06
All	100 to 150	up to 12	2.46E-04	4.78E-04	2.39E-04	2.39E-04	2.73E-06	3.72E-06
All	100 to 5000	up to 30	2.88E-03	4.30E-03	7.57E-03	2.15E-03	7.79E-04	9.15E-05
Withers, Mann	50	12	5.38E-03	1.73E-04	9.73E-05	9.73E-05	4.20E-14	6.80E-22
Broughton, Riddick	70	12	3.97E-03	9.36E-05	2.26E-04	2.26E-04	3.56E-08	1.03E-06
Patterson, Ricks	90	12	3.17E-03	5.80E-05	2.41E-04	2.41E-04	1.08E-06	2.86E-06
DH Hill	150	30	2.00E-03	2.17E-05	7.57E-03	2.15E-03	7.79E-04	9.15E-05
Cox	175	12	1.73E-04	1.58E-05	2.17E-04	2.17E-04	5.68E-06	4.49E-06
Dabney	200	24	1.54E-03	1.22E-05	1.49E-03	5.12E-04	1.85E-04	2.87E-05
Hillsborough St.	200	15	1.54E-03	1.22E-05	2.59E-04	2.59E-04	1.77E-05	7.60E-06
Talley, Reynolds	200	12	9.93E-04	1.21E-05	2.05E-04	2.05E-04	8.97E-06	5.10E-06
Carroll, Syme	325	12	1.07E-03	4.61E-06	1.75E-04	1.64E-04	1.51E-05	5.39E-06
North	350	20	8.23E-04	3.99E-06	4.90E-04	1.76E-04	6.00E-05	1.00E-05
MAXIMUM			***8.54E-03***	***4.30E-03***	***7.57E-03***	***2.15E-03***	***7.79E-04***	***9.15E-05***

[X/Q] value analysis notes:

- Site boundary is located approximately 30 m away from the exhaust stack.
- Closest buildings outside the site boundary are 50 m away (Withers, Mann).
- Closest residential areas are 200 m away (Hillsborough St)
- Student dormitories are 325 m away (Carroll, Syme, North).
- Most buildings are three stories in height.
- DH Hill library is the tallest building near the facility at 150 m away and 30 m high.
- Maximum [X/Q] values are associated with occupied locations that are elevated or closer to the release point from the 30 m reactor stack. Ground level [X/Q] have lower values.
- There are no occupied areas at distances, x, less than 150 m at a height, z, greater than 12 m.

Table 14-11: Calculation 5 - U-235 radiation doses (TEDE) for public areas at the limiting fission rate of 9.6×10^9 f/s

Building or Location	Dist. x (m)	Height z (m)	Eq. 8-4 Fumign. Expos. 24 h (rem)	Eq. 8-4 Calm Wind Expos. 24 h (rem)	Eq. 8-4 GPM Expos. 24 h (rem)	Eq. 8-4 Fumign. Expos. 2 h (rem)	Eq. 8-4 Fumign. Expos. 24 h (rem)	Eq. 8-4 GPM Expos. 96 h (rem)	Eq. 8-4 GPM Expos. 520 h (rem)
All	30 to 100	up to 12	2.73E-04	1.49E-05	7.39E-06	2.22E-04	3.00E-03	1.94E-07	1.19E-05
All	100 to 150	up to 12	7.86E-06	1.53E-05	7.65E-06	6.39E-06	8.64E-05	3.82E-06	2.83E-05
All	100 to 5000	up to 30	9.21E-05	1.38E-04	6.88E-05	7.49E-05	1.01E-03	1.09E-03	6.96E-04
Withers, Mann	50	12	1.72E-04	5.55E-06	3.11E-06	1.40E-04	1.89E-03	5.88E-14	5.17E-21
Broughton, Riddick	70	12	1.27E-04	3.00E-06	7.23E-06	1.03E-04	1.39E-03	4.98E-08	7.83E-06
Patterson, Ricks	90	12	1.01E-04	1.86E-06	7.71E-06	8.24E-05	1.11E-03	1.51E-06	2.18E-05
DH Hill	150	30	6.39E-05	6.96E-07	6.88E-05	5.20E-05	7.03E-04	1.09E-03	6.96E-04
Cox	175	12	5.53E-06	5.07E-07	6.94E-06	4.50E-06	6.08E-05	7.95E-06	3.42E-05
Dabney	200	24	4.92E-05	3.92E-07	1.64E-05	4.00E-05	5.41E-04	2.59E-04	2.18E-04
Hillsbrgh.St.	200	15	4.92E-05	3.92E-07	8.29E-06	4.00E-05	5.41E-04	2.48E-05	5.78E-05
Talley, Reynolds	200	12	3.17E-05	3.88E-07	6.56E-06	2.58E-05	3.49E-04	1.26E-05	3.88E-05
Carroll, Syme	325	12	3.42E-05	1.48E-07	5.25E-06	2.78E-05	3.76E-04	2.11E-05	4.10E-05
North	350	20	2.63E-05	1.28E-07	5.63E-06	2.14E-05	2.89E-04	8.40E-05	7.61E-05
MAXIMUM			***2.73E-04***	***1.38E-04***	***6.88E-05***	***2.22E-04***	***3.00E-03***	***1.09E-03***	***6.96E-04***
RELEASE			*Accident*	*Accident*	*Accident*	*Accident*	*Vented*	*Vented*	*Vented*

U-235 TEDE analysis notes:

Eq. 8-4 for different locations at a given exposure time varies by the ratio of {X/Q] values; i.e. a different [X/Q] is used in Eq. 7-3 or Eq. 7-4 to calculate the time integrated exposure, Ψ_P. Ψ_P is then used in Eq. 8-4 to calculate the public TEDE. Therefore, to determine the TEDE at a specific location, the maximum TEDE may be multiplied by the ratio of the [X/Q] used for specific location under the listed weather conditions at a given exposure time to the maximum [X/Q] used for the same exposure time:

Maximum rem at 24 hours for accidental release using GPM: 6.88×10^{-5} rem = 2.73×10^{-4} rem (2.15×10^{-3} / 8.54×10^{-3})

Talley, Reynolds at 520 hours for vented experiment: 3.88×10^{-5} rem = 6.96×10^{-4} rem (5.10×10^{-6} / 9.15×10^{-5})

Table 14-12: Calculation 5 - Pu-239 radiation doses (TEDE) for other public areas

Building or Location	Dist. x (m)	Height z (m)	Eq. 8-4 Fumign. Expos. 24 h (rem)	Eq. 8-4 Calm Wind Expos. 24 h (rem)	Eq. 8-4 GPM Expos. 24 h (rem)	Eq. 8-4 Fumign. Expos. 2 h (rem)	Eq. 8-4 Fumign. Expos. 24 h (rem)	Eq. 8-4 GPM Expos. 96 h (rem)	Eq. 8-4 GPM Expos. 520 h (rem)
All	30 to 100	up to 12	3.00E-04	1.63E-05	8.10E-06	2.42E-04	2.05E-03	1.33E-07	8.12E-06
All	100 to 150	up to 12	8.64E-06	1.68E-05	8.38E-06	6.97E-06	5.91E-05	2.62E-06	1.94E-05
All	100 to 5000	up to 30	1.01E-04	1.51E-04	7.54E-05	8.16E-05	6.91E-04	7.48E-04	4.76E-04
Withers, Mann	50	12	1.89E-04	6.08E-06	3.41E-06	1.52E-04	1.29E-03	4.03E-14	3.54E-21
Broughton, Riddick	70	12	1.39E-04	3.29E-06	7.93E-06	1.12E-04	9.53E-04	3.42E-08	5.36E-06
Patterson, Ricks	90	12	1.11E-04	2.04E-06	8.45E-06	8.98E-05	7.61E-04	1.04E-06	1.49E-05
DH Hill	150	30	7.03E-05	7.62E-07	7.54E-05	5.67E-05	4.80E-04	7.48E-04	4.76E-04
Cox	175	12	6.08E-06	5.55E-07	7.61E-06	4.90E-06	4.15E-05	5.45E-06	2.34E-05
Dabney	200	24	5.41E-05	4.28E-07	1.80E-05	4.36E-05	3.70E-04	1.78E-04	1.49E-04
Hillsbrgh.St	200	15	5.41E-05	4.28E-07	9.08E-06	4.36E-05	3.70E-04	1.70E-05	3.95E-05
Talley, Reynolds	200	12	3.49E-05	4.25E-07	7.19E-06	2.81E-05	2.38E-04	8.61E-06	2.65E-05
Carroll, Syme	325	12	3.76E-05	1.62E-07	5.75E-06	3.03E-05	2.57E-04	1.45E-05	2.80E-05
North	350	20	2.89E-05	1.40E-07	6.17E-06	2.33E-05	1.98E-04	5.76E-05	5.20E-05
MAXIMUM			***3.00E-04***	***1.51E-04***	***7.54E-05***	***2.42E-04***	***2.05E-03***	***7.48E-04***	***4.76E-04***
RELEASE			***Accident***	***Accident***	***Accident***	***Accident***	***Vented***	***Vented***	***Vented***

CALCULATION 6: Activity for Radionuclides Listed in 10 CFR Part 37 as Quantities of Concern

From Calculations 1 through 5, the limiting fission rate for a release from a fueled experiment was determined to be 9.6 x10^9 f/s. This gives a total number of fissions of 1.8x10^{16} for an irradiation time of 520 h.

Tables 14-13 and 14-14 below give the 10 CFR Part 37 Category 2 Limit Fractions for U-235 and Pu-239 irradiation at 9.6x10^9 f/s for 520 h.

At 1.8 x10^{16} total fissions, activities for radionuclides listed in 10 CFR Part 37 were calculated to be approximately 6.0 x10^{-7} times the activity limits for Catergory 2 Quantities of Concern. The total activity for Sr-90, Cs-137, and Pm-147 is approximately 123 μCi.

Table 14-13: Calculation 6 – 10 CFR Part 37 Category 2 Limit Fraction for U-235 irradiation at 9.6x10^9 f/s for 520 h

Nuclide	Half-life (s)	Decay Constant (1/s)	Thermal Fission Yield %	Non-Thermal Fission Yield %	Eq. 2-1 Activity (Ci)	10 CFR Part 37 Category 2 Limit (Ci)	Fraction of Limit
Sr90	9.07E+08	7.65E-10	5.87	5.60	2.15E-05	3.70E+02	5.82E-08
Cs137	9.47E+08	7.32E-10	3.25	3.76	1.19E-05	2.70E+01	4.42E-07
Pm147	8.26E+07	8.39E-09	2.25	2.14	8.98E-05	1.08E+04	8.32E-09
						Total Limit Fraction =	***5.09E-07***

Table 14-14: Calculation 6 - 10CFR Part 37 Category 2 Limit Fractions for Pu-239 irradiation at $9.6x10^9$ f/s for 520 h

Nuclide	Half-life (s)	Decay Constant (1/s)	Thermal Fission Yield %	Non-Thermal Fission Yield %	Eq. 2-1 Activity (Ci)	10 CFR Part 37 Category 2 Limit (Ci)	Fraction of Limit
Sr90	9.07E+08	7.65E-10	2.15	2.07	7.92E-06	3.70E+02	2.14E-08
Cs137	9.47E+08	7.32E-10	4.28	4.66	1.55E-05	2.70E+01	5.75E-07
Pm147	8.26E+07	8.39E-09	2.01	1.99	8.10E-05	1.08E+04	7.50E-09
						Total Limit Fraction =	***6.04E-07***

CALCULATION 7: External Dose from the Reactor Building, Overhead Plume, and Reactor Stack

Tables 14-15 and 14-16 below give the U-235 and Pu-239 Source Terms at a fission rate of $9.6x10^9$ f/s.

Tables 14-17 through 14-19 give the external dose rates resulting from U-235 at a fission rate of $9.6x10^9$ f/s.

Table 14-20 gives the combined external doses for Pu-239 at a fission rate of $9.6x10^9$ f/s for comparison.

Figures 14-1 and 14-2 illustrate the Microshield model used for the overhead plume and reactor building.

The dose rates from U-235 were calculated to be higher than those from Pu-239 and therefore U-235 is limiting.

External dose from the reactor building, overhead plume, and reactor stack were calculated to give $2.4x10^{-5}$ rem or less to publicly occupied areas outside the reactor building. Most of the dose is associated with the reactor building.

Occupants inside Burlington labs would be evacuated within 15 minutes to areas outside the site boundary. Evacuation time is based on the time for the reactor staff to exit the reactor building and verify personnel within the building have evacuated. This gives a dose of less than $1.4x10^{-5}$ rem based on a minimal distance of 1 m to 10 m occupied for 15 minutes.

University personnel notify reactor staff if roof top access is being made. Reactor facility procedures require the reactor staff to clear the roof top if an evacuation alarm occurs. Initial dose rate on the roof top was calculated to be $8.6x10^{-4}$ rem/h. Evacuation time is estimated as being less than 15 minutes based on the time for the reactor staff to exit the reactor building and notify personnel on the roof top. This gives a dose of approximately $2.2x10^{-4}$ rem or less to roof top occupants.

Table 14-15: Calculation 7 - U-235 Source Term at 9.6×10^9 f/s

Nuclide	Eq. 5-1 Initial Conc. C(0) (µCi/ml)	Eq. 5-2 2 h Average Reactor Conc. <C> (µCi/ml)	Eq. 5-2 24 h Average Reactor Conc. <C> (µCi/ml)	Eq. 5-2 2 h Average Stack Conc. <C> (µCi/ml)	Eq. 8-1 2 h Average Stack Line (Ci)	Eq. 8-2 2 h Average Overhead Line Calm Wind (Ci)	Eq. 5-2 24 h Average Stack Conc. <C> (µCi/ml)	Eq. 8-1 24 h Average Stack Line (Ci)	Eq. 8-2 24 h Average Overhead Line Calm Wind (Ci)
83mKr	6.28E-07	3.07E-07	3.17E-08	3.07E-07	1.21E-06	1.74E-05	3.17E-08	1.24E-07	1.79E-06
85mKr	1.50E-06	8.71E-07	1.03E-07	8.71E-07	3.42E-06	4.93E-05	1.03E-07	4.05E-07	5.84E-06
85Kr	3.30E-07	2.17E-07	3.03E-08	2.17E-07	8.51E-07	1.23E-05	3.03E-08	1.19E-07	1.72E-06
87Kr	2.94E-06	1.27E-06	1.23E-07	1.27E-06	5.00E-06	7.22E-05	1.23E-07	4.82E-07	6.96E-06
88Kr	4.06E-06	2.19E-06	2.43E-07	2.19E-06	8.60E-06	1.24E-04	2.43E-07	9.53E-07	1.37E-05
89Kr	5.06E-06	1.85E-07	1.54E-08	1.85E-07	7.27E-07	1.05E-05	1.54E-08	6.06E-08	8.74E-07
131mXe	4.53E-08	2.98E-08	4.15E-09	2.98E-08	1.17E-07	1.69E-06	4.15E-09	1.63E-08	2.35E-07
133mXe	2.20E-07	1.43E-07	1.97E-08	1.43E-07	5.61E-07	8.10E-06	1.97E-08	7.72E-08	1.11E-06
133Xe	7.72E-06	5.06E-06	7.02E-07	5.06E-06	1.98E-05	2.86E-04	7.02E-07	2.75E-06	3.97E-05
135mXe	1.31E-06	2.06E-07	1.72E-08	2.06E-07	8.09E-07	1.17E-05	1.72E-08	6.76E-08	9.75E-07
135Xe	7.55E-06	4.66E-06	5.94E-07	4.66E-06	1.83E-05	2.64E-04	5.94E-07	2.33E-06	3.36E-05
137Xe	7.02E-06	3.10E-07	2.58E-08	3.10E-07	1.22E-06	1.75E-05	2.58E-08	1.01E-07	1.46E-06
138Xe	7.18E-06	1.05E-06	8.79E-08	1.05E-06	4.14E-06	5.97E-05	8.79E-08	3.45E-07	4.98E-06
131I	3.42E-06	2.24E-06	3.12E-07	2.24E-07	8.79E-07	1.27E-05	3.12E-08	1.22E-07	1.77E-06
132I	5.06E-06	2.61E-06	2.79E-07	2.61E-07	1.02E-06	4.93E-06	2.79E-08	1.10E-07	1.58E-06
133I	7.72E-06	4.94E-06	6.61E-07	4.94E-07	1.94E-06	2.80E-05	6.61E-08	2.60E-07	3.75E-06
134I	8.97E-06	3.31E-06	3.01E-07	3.31E-07	1.30E-06	1.87E-05	3.01E-08	1.18E-07	1.70E-06
135I	7.24E-06	4.36E-06	5.40E-07	4.36E-07	1.71E-06	2.47E-05	5.40E-08	2.12E-07	3.06E-06
83Br	6.31E-07	3.29E-07	3.54E-08	3.29E-08	1.29E-07	1.86E-06	3.54E-09	1.39E-08	2.01E-07
84Br	1.13E-06	3.10E-07	2.66E-08	3.10E-08	1.22E-07	1.76E-06	2.66E-09	1.05E-08	1.51E-07

Supporting Calculations

C(0) = [A(∞)/V] where A(∞) is taken from Eq. 2-1 as shown in Table 14-1 for U-235

Average Release Concentration (Reference Eq. 5-1 and 5-2):

$< C > = \int C(0)\, e^{-kt} dt = C(0)\, [\, (1- e^{-kT}) / (kT)]$

For Kr-87:

$C(0) = 6.63 \times 10^3$ µCi/ 2.25×10^9 ml = 2.94×10^{-6} µCi/ml

2 h <C> = (2.94×10^{-6} µCi/ml) [(1-exp(-2.77 $\times 10^{-4}$/s*7200s)) / (2.77 $\times 10^{-4}$/s*7200s)]
= 1.27×10^{-6} µCi/ml

Line Activity (Reference Eq. 8-1 and 8-2)

Stack Ci = 3.93 <C> at 24 h for Kr-85 = (3.93)(3.03×10^{-8}) Ci = 1.19×10^{-7} Ci

Overhead Line Ci = 56.6 <C> at 2 h for Xe-138 = (56.6)(1.05×10^{-6}) Ci = 5.97×10^{-5} Ci

Table 14-16: Calculation 7 - Pu-239 Source Term at 9.6x10^9 f/s

Nuclide	Eq. 5-1 Initial Conc. C(0) (µCi/ml)	Eq. 5-2 2 h Average Reactor Conc. <C> (µCi/ml)	Eq. 5-2 24 h Average Reactor Conc. <C> (µCi/ml)	Eq. 5-2 2 h Average Stack Conc. <C> (µCi/ml)	Eq. 8-1 2 h Average Stack Line (Ci)	Eq. 8-2 2 h Average Overhead Line Calm Wind (Ci)	Eq. 5-2 24 h Average Stack Conc. <C> (µCi/ml)	Eq. 8-1 24 h Average Stack Line (Ci)	Eq. 8-2 24 h Average Overhead Line Calm Wind (Ci)
83mKr	3.47E-07	1.70E-07	1.75E-08	1.70E-07	6.67E-07	9.63E-06	1.75E-08	6.88E-08	9.92E-07
85mKr	6.58E-07	3.81E-07	4.51E-08	3.81E-07	1.49E-06	2.16E-05	4.51E-08	1.77E-07	2.55E-06
85Kr	1.46E-07	9.60E-08	1.34E-08	9.60E-08	3.77E-07	5.43E-06	1.34E-08	5.27E-08	7.60E-07
87Kr	1.15E-06	4.99E-07	4.81E-08	4.99E-07	1.96E-06	2.83E-05	4.81E-08	1.89E-07	2.73E-06
88Kr	1.47E-06	7.94E-07	8.79E-08	7.94E-07	3.11E-06	4.49E-05	8.79E-08	3.45E-07	4.98E-06
89Kr	1.67E-06	6.13E-08	5.11E-09	6.13E-08	2.41E-07	3.47E-06	5.11E-09	2.01E-08	2.89E-07
131mXe	4.90E-08	3.22E-08	4.48E-09	3.22E-08	1.26E-07	1.82E-06	4.48E-09	1.76E-08	2.54E-07
133mXe	2.71E-07	1.76E-07	2.42E-08	1.76E-07	6.91E-07	9.97E-06	2.42E-08	9.49E-08	1.37E-06
133Xe	8.08E-06	5.29E-06	7.34E-07	5.29E-06	2.08E-05	2.99E-04	7.34E-07	2.88E-06	4.16E-05
135mXe	2.19E-06	3.44E-07	2.87E-08	3.44E-07	1.35E-06	1.95E-05	2.87E-08	1.13E-07	1.63E-06
135Xe	8.75E-06	5.40E-06	6.89E-07	5.40E-06	2.12E-05	3.06E-04	6.89E-07	2.70E-06	3.90E-05
137Xe	6.81E-06	3.00E-07	2.50E-08	3.00E-07	1.18E-06	1.70E-05	2.50E-08	9.82E-08	1.42E-06
138Xe	5.83E-06	8.56E-07	7.14E-08	8.56E-07	3.36E-06	4.85E-05	7.14E-08	2.80E-07	4.05E-06
131I	4.45E-06	2.92E-06	4.06E-07	2.92E-07	1.15E-06	1.65E-05	4.06E-08	1.59E-07	2.30E-06
132I	6.19E-06	3.20E-06	3.42E-07	3.20E-07	1.25E-06	1.81E-05	3.42E-08	1.34E-07	1.94E-06
133I	8.02E-06	5.13E-06	6.87E-07	5.13E-07	2.01E-06	2.91E-05	6.87E-08	2.70E-07	3.89E-06
134I	8.46E-06	3.12E-06	2.83E-07	3.12E-07	1.22E-06	1.77E-05	2.83E-08	1.11E-07	1.61E-06
135I	7.41E-06	4.46E-06	5.53E-07	4.46E-07	1.75E-06	2.53E-05	5.53E-08	2.17E-07	3.13E-06
83Br	3.47E-07	1.81E-07	1.95E-08	1.81E-08	7.10E-08	1.03E-06	1.95E-09	7.66E-09	1.10E-07
84Br	5.04E-07	1.39E-07	1.19E-08	1.39E-08	5.45E-08	7.87E-07	1.19E-09	4.68E-09	6.76E-08

Table 14-17: Calculation 7 - External dose rates using Microshield and calculated dose from the reactor building for U-235 at 9.6x10^9 f/s

Distance (m)	Initial (rem / h)	2 h (rem / h)	24 h (rem / h)	2h (rem)	24 h (rem)
1	5.58E-05	2.03E-05	2.19E-06	4.07E-05	5.25E-05
10	2.39E-05	8.82E-06	9.51E-07	1.76E-05	2.28E-05
20	9.96E-06	3.69E-06	3.96E-07	7.38E-06	9.50E-06
30	5.13E-06	1.91E-06	2.05E-07	3.81E-06	4.92E-06
40	3.06E-06	1.13E-06	1.22E-07	2.26E-06	2.92E-06
50	1.97E-06	7.29E-07	7.86E-08	1.46E-06	1.89E-06
Roof	8.55E-04				

Notes:

- 24 h dose at 30 m = 4.92×10^{-6} rem = (24 h) (2.05×10^{-7} rem/h)
- The 1 m distance is associated with offices in Burlington labs. 10 m to 30 m are associated with distances to the site boundary. Nearby buildings are located at 30 m to 50 m.

Table 14-18: Calculation 7 - External dose rates from Microshield and calculated dose from overhead plume and reactor stack for U-235 at $9.6x10^9$ f/s

Location x,y,z (m)	2 h Average Stack Line (rem / h)	2 h Average Overhead Line (rem / h)	2 h Average Stack and Plume (rem)	24 h Average Stack Line (rem / h)	24 h Average Overhead Line (rem / h)	24 h Average Stack and Plume (rem)
30,0,0	1.59E-08	1.92E-07	4.16E-07	1.73E-09	2.09E-08	5.42E-07
40,0,0	1.01E-08	2.01E-07	4.22E-07	1.10E-09	2.18E-08	5.48E-07
50,0,0	6.84E-09	2.03E-07	4.20E-07	7.41E-10	2.21E-08	5.47E-07
10,0,12	1.34E-07	3.12E-07	8.92E-07	1.45E-08	3.39E-08	1.16E-06
20,0,12	4.38E-08	3.66E-07	8.20E-07	4.74E-09	3.96E-08	1.06E-06
30,0,12	2.10E-08	3.93E-07	8.28E-07	2.28E-09	4.26E-08	1.08E-06
40,0,12	1.21E-08	4.05E-07	8.34E-07	1.32E-09	4.41E-08	1.09E-06
50,0,12	7.74E-09	4.11E-07	8.37E-07	8.37E-10	4.44E-08	1.09E-06

Table 14-19: Calculation 7 - Combined external doses (sum of reactor building, overhead plume, and stack) for U-235 at $3.2x10^9$ f/s

Location x,y,z (m)	External Dose in 2 h (rem)	External Dose in 24 h (rem)
10,0,12	1.9E-05	2.4E-05
20,0,12	8.2E-06	1.1E-05
30,0,12	4.6E-06	6.0E-06
40,0,12	3.1E-06	4.0E-06
50,0,12	2.3E-06	3.0E-06

Note: 2 h dose at 50 m = $2.3x10^{-6}$ rem = ($8.34x10^{-7}$ + $1.46x10^{-6}$) rem

Table 14-20: Calculation 7 - Combined external doses (sum of reactor building, overhead plume, and stack) for Pu-239 at $3.2x10^9$ f/s

Location x,y,z (m)	External Dose in 2 h (rem)	External Dose in 24 h (rem)
10,0,12	1.6E-05	2.1E-05
20,0,12	6.9E-06	9.0E-06
30,0,12	3.9E-06	5.0E-06
40,0,12	2.5E-06	3.3E-06
50,0,12	1.8E-06	2.4E-06

Microshield models used for the overhead plume and reactor building are illustrated below.

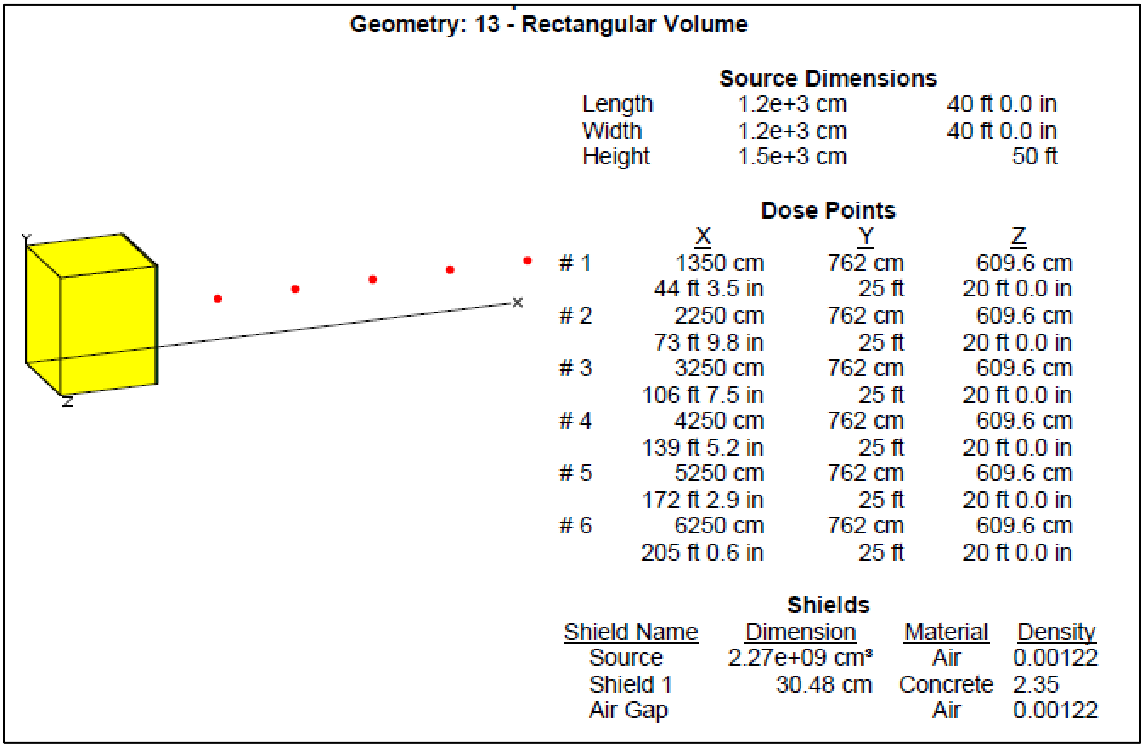

Figure 14-1: Rectangular volume geometry (reactor building with shield is shown)

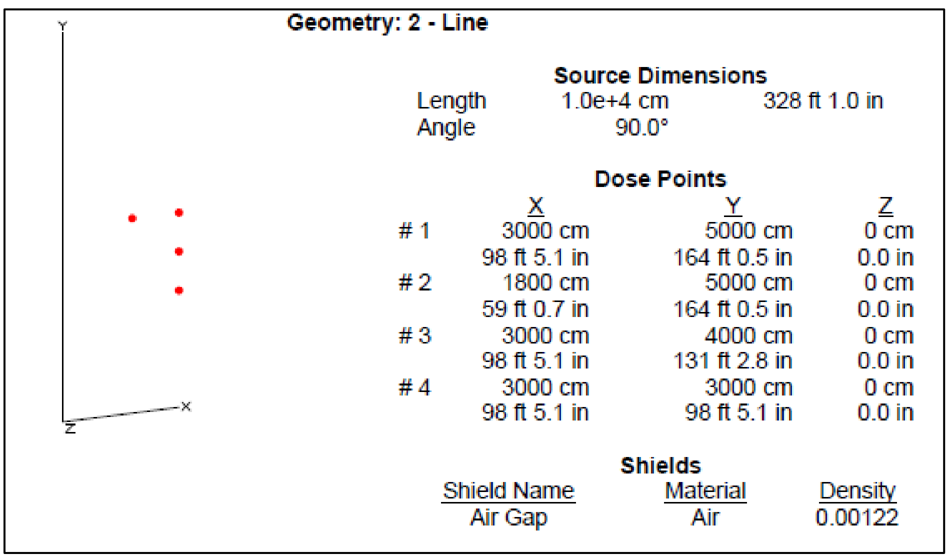

Figure 14-2: Line source geometry (overhead plume is shown)

CALCULATION 8: External dose rates from beam tube exhaust duct

Tables 14-21 and 14-22 give the beam tube exhaust duct activity and total dose resulting from U-235 with a fission rate of 9.6×10^9 f/s with a 520 hour exposure.

Tables 14-23 and 14-24 give the beam tube exhaust duct activity and total dose resulting from Pu-239 with a fission rate of 9.6×10^9 f/s with a 520 hour exposure.

Results for U-235 were slightly higher than those for Pu-239. The total activity in the beam tube exhaust duct at a fission rate of 9.6×10^9 f/s for a vented experiment is 274 μCi for U-235 and 265 μCi for Pu-239.

External dose rates for occupied areas (3 m or greater) from the beam tube exhaust duct are 7.7×10^{-6} rem/h or less for U-235 and 7.3×10^{-6} rem/h or less for Pu-239. The total dose for 520 h exposure is 4.0×10^{-3} rem or less for U-235 and for Pu-239.

Table 14-21: Calculation 8 - Beam Tube Exhaust Duct Activity resulting from U-235 at 9.6×10^9 f/s

Nuclide	Eq. 5-8 Decayed, Filtered Vented Release Rate q, (μCi/s)	Eq. 8-3 Beam Tube Duct A(d), (μCi)
83mKr	6.51E-01	2.61E+00
85mKr	1.74E+00	6.96E+00
85Kr	4.12E-01	1.65E+00
87Kr	2.80E+00	1.12E+01
88Kr	4.49E+00	1.80E+01
89Kr	8.59E-03	3.44E-02
131mXe	5.66E-02	2.26E-01
133mXe	2.73E-01	1.09E+00
133Xe	9.63E+00	3.85E+01
135mXe	4.21E-01	1.68E+00
135Xe	9.08E+00	3.63E+01
137Xe	3.80E-02	1.52E-01
138Xe	2.05E+00	8.22E+00
131I	4.26E+00	1.70E+01
132I	5.44E+00	2.17E+01
133I	9.49E+00	3.80E+01
134I	7.55E+00	3.02E+01
135I	8.58E+00	3.43E+01
83Br	6.83E-01	2.73E+00
84Br	7.31E-01	2.93E+00
	6.84E+01	2.74E+02

Supporting Calculations:

Ventilation Exhaust Activity (Reference Eq. 5-8 and 8-3):

$A(d) = 4\ q = 4\ [A(\infty) / w]\ \ p\ [\exp(-\lambda w/p)]$ where $t = w/p$

Kr-87: 2.8 μCi = (4s) (6.63×10^3 μCi / 9.0×10^4 ml)(50 ml/s)[exp(-1.52×10^{-4} x 9.0×10^4/50)], or 2.8×10^{-6} Ci

Using a line source geometry in Microshield with the source term [A(d)] from Table 14-21 gives the dose rates in Table 14-22:

Table 14-22: Calculation 8 - Beam Tube Exhaust Duct Dose Rates from Microshield and Calculated Dose resulting from U-235 at $9.6x10^9$ f/s and 520 hour exposure

Distance (m)	Dose Rate (rem/h)	Total Dose (rem)
1	2.80E-05	1.5E-02
2	1.28E-05	6.6E-03
3	7.71E-06	4.0E-03
4	5.25E-06	2.7E-03
5	3.81E-06	2.0E-03
6	2.89E-06	1.5E-03
7	2.27E-06	1.2E-03
8	1.82E-06	9.5E-04
9	1.49E-06	7.7E-04
10	1.24E-06	6.4E-04

Table 14-23: Calculation 8 - Beam Tube Exhaust Duct Activity resulting from Pu-239 at $9.6x10^9$ f/s

Nuclide	Eq. 5-8 Decayed, Filtered Vented Release Rate q, (μCi/s)	Eq. 8-3 Beam Tube Duct A(d), (μCi)
83mKr	3.60E-01	1.44E+00
85mKr	7.61E-01	3.04E+00
85Kr	1.82E-01	7.30E-01
87Kr	1.10E+00	4.39E+00
88Kr	1.63E+00	6.51E+00
89Kr	2.84E-03	1.14E-02
131mXe	6.12E-02	2.45E-01
133mXe	3.36E-01	1.34E+00
133Xe	1.01E+01	4.03E+01
135mXe	7.02E-01	2.81E+00
135Xe	1.05E+01	4.21E+01
137Xe	3.68E-02	1.47E-01
138Xe	1.67E+00	6.68E+00
131I	5.56E+00	2.22E+01
132I	6.66E+00	2.66E+01
133I	9.86E+00	3.95E+01
134I	7.12E+00	2.85E+01
135I	8.79E+00	3.52E+01
83Br	3.76E-01	1.50E+00
84Br	3.28E-01	1.31E+00
Total	6.61E+01	2.65E+02

Table 14-24: Calculation 8 - Beam Tube Exhaust Duct Dose Rates from Microshield and Calculated Dose resulting from Pu-239 at 9.6×10^9 f/s and 520 hour exposure

Distance (m)	Dose Rate (rem/h)	Total Dose (rem)
1	2.65E-05	1.4E-02
2	1.21E-05	6.3E-03
3	7.30E-06	3.8E-03
4	4.98E-06	2.6E-03
5	3.60E-06	1.9E-03
6	2.74E-06	1.4E-03
7	2.15E-03	1.1E+00
8	1.73E-06	9.0E-04
9	1.41E-06	7.3E-04
10	1.18E-06	6.1E-04

CALCULATION 9: Irradiation of Uranium

Table 14-25 gives the parameter values for experiments containing uranium.

Table 14-26 gives the time integrated exposures and radiation doses

Table 14-27 gives the dose summary for experiments containing uranium.

At the fluence rates used in this analysis, a fission rate of 1.9×10^6 f/s is equivalent to 9.8×10^{-7} g of U-235. Results indicate 1.9×10^6 f/s is limiting based on the dose criteria of one percent (1%) of the annual radiation dose limits given in 10 CFR Part 20 for experiments containing uranium.

For experiments containing uranium, a fission rate of 1.9×10^6 f/s has a TEDE of 1.0×10^{-3} rem or less to personnel within the reactor building and a TEDE of 1.0×10^{-5} rem or less to members of the public outside the reactor building. Experiments with uranium equal to or greater than 1.9×10^6 f/s or 1.6×10^{11} fissions are therefore defined as a fueled experiment.

Table 14-25: Calculation 9 – Parameter values for experiments containing uranium

PARAMETER VALUES					
Parameter	**Value**	**Unit**	**Parameter**	**Value**	**Units**
Nuclide	*U-235*		Target atoms, N	*2.53E+15*	atoms
Mass	*9.86E-07*	g	Thermal fission rate	*1.48E+06*	f/s
Mass Number, A	*235*	g/mol	Non-thermal fission rate	*4.33E+05*	f/s
Sigma thermal	*585*	b	Total fission rate	*1.91E+06*	f/s
Sigma non-thermal	*571*	b	Reactor volume	*2.25E+09*	ml
X/Q	*8.54E-03*	s/m^3	F normal	*0.883*	m^3/s
Thermal flux	*1.00E+12*	cm^2/s	v normal	*3.92E-04*	1/s
Non-thermal flux	*3.00E+11*	cm^2/s	NG reactor correction	*0.1*	
Irradiation time	*520*	h	Fissions	*1.65E11*	
Public Hours	*24*	h			
Reactor exposure time	*24*	h			

ISOTOPIC DATA

Nuclide	Half-Life (sec)	Decay Constant (1/s)	Cumulative Yield % Thermal Fission	Cumulative Yield % Non-Thermal Fission	Eq. 2-1 Saturation Activity (μCi)
83mKr	6.70E+03	1.04E-04	***5.36E-01***	***5.75E-01***	3.88E-02
85mKr	1.61E+04	4.30E-05	***1.29E+00***	***1.36E+00***	9.30E-02
85Kr	3.39E+08	2.05E-09	***2.83E-01***	***2.96E-01***	2.04E-02
87Kr	4.57E+03	1.52E-04	***2.56E+00***	***2.54E+00***	1.82E-01
88Kr	1.02E+04	6.78E-05	***3.55E+00***	***3.43E+00***	2.51E-01
89Kr	1.89E+02	3.67E-03	***4.51E+00***	***3.97E+00***	3.13E-01
131mXe	1.03E+06	6.74E-07	***4.05E-02***	***3.54E-02***	2.80E-03
133mXe	1.89E+05	3.66E-06	***1.89E-01***	***1.97E-01***	1.36E-02
133Xe	4.53E+05	1.53E-06	***6.70E+00***	***6.71E+00***	4.77E-01
135mXe	9.18E+02	7.55E-04	***1.10E+00***	***1.26E+00***	8.10E-02
135Xe	3.28E+04	2.12E-05	***6.54E+00***	***6.58E+00***	4.67E-01
137Xe	2.29E+02	3.02E-03	***6.13E+00***	***5.98E+00***	4.34E-01
138Xe	8.46E+02	8.19E-04	***6.30E+00***	***6.00E+00***	4.44E-01
131I	6.93E+05	1.00E-06	***2.89E+00***	***3.22E+00***	2.11E-01
132I	8.26E+03	8.39E-05	***4.31E+00***	***4.66E+00***	3.13E-01
133I	7.49E+04	9.26E-06	***6.70E+00***	***6.70E+00***	4.77E-01
134I	3.16E+03	2.20E-04	***7.83E+00***	***7.63E+00***	5.55E-01
135I	2.37E+04	2.93E-05	***6.28E+00***	***6.27E+00***	4.47E-01
83Br	8.64E+03	8.02E-05	***5.40E-01***	***5.76E-01***	3.90E-02
84Br	1.91E+03	3.63E-04	***9.67E-01***	***1.01E+00***	6.96E-02

Table 14-26: Calculation 9 – Time Integrated Exposures and Doses from Irradiation of Uranium at 1.9 E6 f/s

Nuclide	Eq. 7-1* Time Integrated Exposure-Normal Reactor (μCi-h/ml)	Eq. 7-5 Time Integrated Exposure-Normal Public (μCi-h/ml)	Effective Inhalation DCF (rem per μCi-h/ml)	Thyroid Inhalation DCF (rem per μCi-h/ml)	Submersion DCF (rem per μCi-h/ml)	Eq. 8-4 Reactor Normal TEDE (rem)	Eq. 8-4 Reactor Normal Thyroid Dose (rem)	Eq. 8-4 Public Normal TEDE (rem)
83mKr	4.14E-10	3.12E-12			1.52E-02	6.28E-13	6.28E-13	4.73E-14
85mKr	9.92E-10	7.48E-12			1.10E+02	1.09E-08	1.09E-08	8.24E-10
85Kr	2.17E-10	1.64E-12			1.74E+00	3.78E-11	3.78E-11	2.85E-12
87Kr	1.94E-09	1.46E-11			5.25E+02	1.02E-07	1.02E-07	7.69E-09
88Kr	2.68E-09	2.02E-11			1.33E+03	3.57E-07	3.57E-07	2.69E-08
89Kr	3.33E-09	2.51E-11			1.20E+03	4.00E-07	4.00E-07	3.02E-08
131mXe	2.99E-11	2.25E-13			5.48E+00	1.64E-11	1.64E-11	1.23E-12
133mXe	1.45E-10	1.09E-12			1.99E+01	2.89E-10	2.89E-10	2.18E-11
133Xe	5.09E-09	3.84E-11			2.25E+01	1.14E-08	1.14E-08	8.62E-10
135mXe	8.64E-10	6.51E-12			2.79E+02	2.41E-08	2.41E-08	1.81E-09
135Xe	4.98E-09	3.75E-11			1.73E+02	8.62E-08	8.62E-08	6.49E-09
137Xe	4.63E-09	3.49E-11			1.10E+02	5.10E-08	5.10E-08	3.84E-09
138Xe	4.73E-09	3.57E-11			7.10E+02	3.36E-07	3.36E-07	2.54E-08
131I	2.25E-09	1.70E-11	3.95E+04	1.30E+06	2.42E+02	8.90E-05	2.92E-03	6.74E-07
132I	3.34E-09	2.51E-11	4.57E+02	7.73E+03	1.49E+03	2.02E-06	2.63E-05	4.90E-08
133I	5.09E-09	3.84E-11	7.02E+03	2.16E+05	3.92E+02	3.59E-05	1.10E-03	2.84E-07
134I	5.91E-09	4.46E-11	1.58E+02	1.28E+03	1.73E+03	1.96E-06	8.59E-06	8.42E-08
135I	4.77E-09	3.60E-11	1.47E+03	3.76E+04	1.06E+03	7.54E-06	1.80E-04	9.12E-08
83Br	4.16E-10	3.14E-12	1.03E+02		5.09E+00	4.33E-08	2.12E-10	3.40E-10
84Br	7.42E-10	5.59E-12	1.01E+02		1.25E+03	1.68E-07	9.30E-08	7.57E-09

***Note:** Eq 5-1 is used in Eq 7-1 for <C>, or <C> = C(0) giving $\psi = [A(\infty) / V]\ T$

Table 14-27: Calculation 9 – Dose Summary for Experiments containing Uranium at 1.9×10^6 f/s

	Reactor Building	Public Areas
Total TEDE, (rem) =	0.001	9.39E-06
TEDE Limit, (rem) =	0.001	0.001
rem dose per f/s=	5.24E-10	4.91E-12
Eq 9-1 Fission rate limit (f/s) =	***1.91E+06***	*2.04E+08*

From Eq. 9-3, the total number of fissions is 1.6×10^{11} for an irradiation time of 520 hours.

SECTION 15: CONCLUSIONS

Table 15-1 below provides a summary of calculated radiation doses inside and outside of the reactor building resulting from planned vented and accidental releases for fueled experiments performed at the limiting conditions given in the requested technical specifications.

Given the low limits requested for fission rates and total fissions, any doses arising from accidental or planned vented releases from fueled experiments will be more than an order of magnitude below 10 CFR 20 limits.

Table 15-1: Radiation Doses for Fueled Experiments for planned vented and accidental releases, as compared to 10 CFR Part 20 Limits

Release	Nuclide	Reactor Bldg. TEDE (rem)	Reactor Bldg. Thyroid TODE (rem)	Public TEDE (rem)
Vented	U-235	0.004		0.003
Vented	Pu-239	0.0038		0.002
Accidental	U-235	0.02	0.62	0.00027
Accidental	Pu-239	0.025	0.75	0.0003
Dose Limits for Fueled Experiments:		**0.15**	**1.5**	**0.003**
	10 CFR Part 20 Limits:	***5.0*** (1)	***50*** (2)	***0.1*** (3)

(1): TEDE Annual Limit for occupationally exposed radiation worker.
(2): Thyroid TODE for occupational radiation worker
(3): TEDE Annual Limit for member of the public.

Fueled experiments, as analyzed, meet the following criteria:

- As given in Table 15-1 above, radiation dose does not exceed:
  - TEDE of 0.15 rem to occupants inside the reactor building.
  - Thyroid TODE of 1.5 rem to occupants inside the reactor building.
  - TEDE of 0.003 rem in public areas outside the reactor building.
- Emergency action levels as defined in the facility emergency plan are not exceeded.
- Limits for a reportable event as defined in TS 1.2.24 b are not exceeded.
- Fissionable materials are stored as required by TS 5.3 and the facility security plan and radiation protection program.

Based on the analysis for fueled experiments for the assumed conditions and radiation dose criteria, the following conclusions are made:

- The fission rate for fueled experiments is limited to 9.6×10^{9} fissions per second.
- The total number of fissions for fueled experiments is limited to 1.8×10^{16} fissions.
- Individual or mixtures of fissionable material meeting the limiting fission rate and total number of fissions may be used in a fueled experiment.

- Fissionable material mass and experiment fluence rates may be adjusted such that the limiting fission rate is met.

- Uranium with a fission rate greater than 1.9×10^{6} fissions per second or total number of fissions greater than 1.6×10^{11} fissions is defined as a fueled experiment.

- Materials containing Neptunium or Plutonium that undergo fission in an experiment are considered to be fueled experiments.

The following additional controls will be implemented for vented experiments:

- Filtration of particulates and halogens is required. Filters with rated retention greater than 95 percent for particulates and halogens are to be used. Retention for halogens was assumed to be 90 percent.

- A minimum decay time of 0.5 hours (30 minutes) before entering the reactor building ventilation system.

- The maximum experiment exhaust rate is 3 liters per minute.

- The experiment exhaust is well mixed prior to release into the reactor building ventilation system.

- Experiment exhaust flow rate is routed to the ventilation system using the beam tube exhaust and controlled by dedicated equipment with local flow rate indication.

- The experiment exhaust is capable of being isolated.

- The exhaust flow tubing from the experiment to the beam tube exhaust is sealed to prevent leakage into the reactor building free air space.

- Radiation monitoring and flow rate monitoring of the experiment exhaust prior to being routed to the reactor building ventilation system is required to identify the source of the release. The release is monitored for radioactivity with indication locally and in the control room. Local alarm annunciation and alarm indication in the control room is provided.

SECTION 16: REFERENCES

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