NC STATE UNIVERSITY

SAFETY ANALYSIS IN SUPPORT OF FUELED EXPERIMENTS FOR THE NCSU PULSTAR REACTOR

Nuclear Reactor Program

NORTH CAROLINA STATE UNIVERSITY

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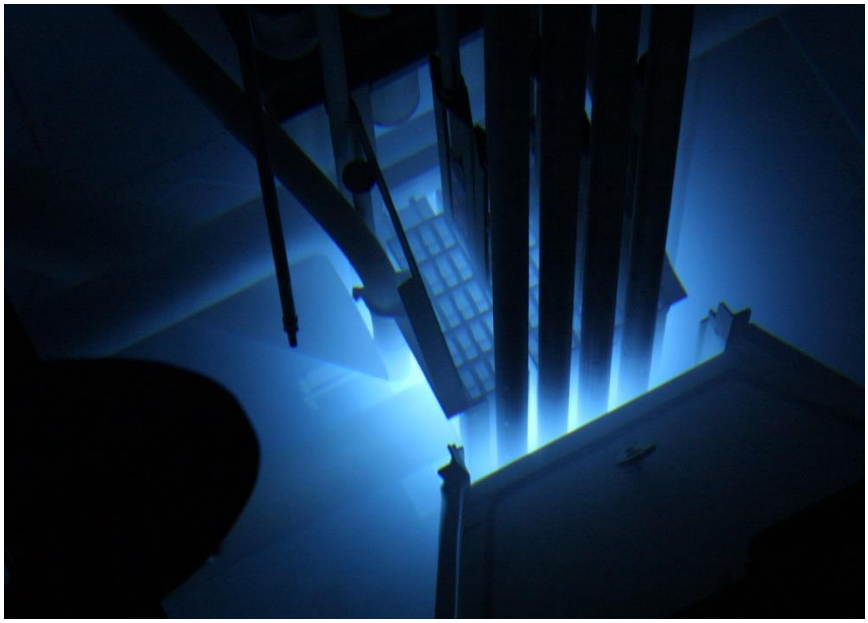

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CONTENTS

EXECUTIVE SUMMARY

The PULSTAR Reactor facility has both present and pending needs to expand its capabilities to perform fueled experiments. Proposed fueled experiments fall into two general classes: 1) vented, and 2) encapsulated. Vented fueled experiments would be designed to permit a limited activity stream of fission gases and halogens to be emitted from fissionable materials under irradiation and analyzed prior to filtering, decay and release via the facility exhaust system. For example, a vented fission gas measurement loop facility is planned for installation in a reactor beamtube which would allow the measurement of fission gas release rates from small samples of fissionable material (e.g. LEU) under active irradiation and high temperature conditions. Encapsulated fueled experiments would involve the sealing of fissionable materials in suitable containers and irradiation to required fluences. For example, an encapsulated fueled experiment is currently proposed to support non-proliferation R&D for LEU fuel materials. Given the increased level of interest in small modular reactors, molten salt reactors, non-proliferation objectives, etc., combined with the existing irradiation and post-irradiation examination (PIE) capabilities of the PULSTAR, additional fueled experiments are likely to be proposed at the PULSTAR reactor facility in the near future.

Both classes of fueled experiments discussed above would require changes to the relevant facility technical specifications (TS) and license possession limits prior to being performed. Given the diverse nature of fueled experiments under consideration, it is desirable to amend the Technical Specifications (TS) and license conditions in such a way as to permit flexibility in the design and performance of these experiments, while simultaneously adhering to a strong safety basis. The primary safety basis proposed is to require that each fueled experiment, whether vented or encapsulated, be limited to a maximum permissible radiation dose less than or equal to a Total Effective Dose Equivalent (TEDE) of 0.003 rem per calendar year in occupied areas outside the reactor building from an accidental or planned release of fission gases and halogens.

The analysis presented in this document provides the methodology, assumptions, data and supporting calculations to ensure that the radiation dose basis is maintained and justifies the requested technical specifications for all fueled experiments. The methodology and equations covering a range of different potential experiments 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 16 provide supporting data, calculations, and conclusions.

A number of experimental configurations for fueled experiments of both the vented and encapsulated classes were evaluated for accidental and planned releases through varying experimental design parameters while maintaining the TEDE at 0.003 rem for all operational and credible failure modes (reference Table 14-31 through Table 14-33). Mixtures of fissionable isotopes typical of different fuel samples were evaluated, with masses, fluxes, fluences, and exhaust decay and flow rates (if applicable) varied while confirming that the TEDE stayed within the basis. Setpoints for vented experiment exhaust gas flow rate and radiation monitors were determined for each experiment configuration, along with that for the existing facility stack radiation monitors, to assure that sufficient active indication and automatic protective actions would be provided to confirm that the TEDE of 0.003 rem is not exceeded.

Section 14 of the analysis provides numerous examples of representative fueled experiments to demonstrate the range of limits that would be applied on a case-by-case basis. In all cases, the safety basis of the radiation dose being less than or equal to a TEDE of 0.003 rem for members of the public in occupied areas outside the reactor building is maintained.

The TS dose requirement of TEDE of 0.003 rem to publicly occupied areas outside the reactor building is 3.3 times lower than the constraint radiation dose given in 10 CFR Part 20 (TEDE of 0.010 rem), and 33 times lower than the annual radiation dose limit for members of the public given in 10 CFR Part 20 (TEDE of 0.1 rem), i.e. only 3 percent. Applying this low radiation dose limit, the associated release would result in activities and doses below twenty percent of the annual occupational radiation dose limits given in 10 CFR Part 20 inside the reactor building, below that for a reportable event given in TS, below an emergency action level requiring activation of the facility emergency plan, and below Category 2 Quantities of Concern activity limits given in 10 CFR Part 37.

Given this low radiation dose and the conservatisms applied as detailed in this analysis, radiation dose to the public from fueled experiments performed at the facility would be minimal and indiscernible from background levels.

As a result of these analyses performed in support of changing the facility technical specifications (TS) and license possession limits to allow for fueled experiments, the following statements are made:

1) The maximum radiation dose from the release of fission gases and halogens produced in a fueled experiment in a calendar year will be less than or equal to a TEDE of 0.003 rem to members of the public in occupied areas outside the reactor building. Activity releases and/or radiation doses do not exceed the limits established for a reportable event or emergency action levels.
2) Amounts of material requested and associated fission product activity does not exceed 10 CFR Part 37 Category 2 limits.
3) Fissionable materials will continue to be stored as required by the R-120 reactor license and Technical Specification, the facility Physical Security Plan and the Radiation Protection Program.
4) No changes are required to the approved facility Emergency Plan, Security Plan, or Radiation Protection Program.

The requested changes to the R-120 reactor license and Technical Specifications (TS) needed to support fueled experiments are therefore as follows:

1) R-120 reactor license Section 2.B.(2) regarding possession limits for fissionable materials to be used in fueled experiments.
2) Technical Specification 1.3.10.f for the definition of a fueled experiment.
3) Technical Specification 3.5 for monitoring of vented fueled experiments.
4) Technical Specification 3.8 for limiting conditions for operations for fueled experiments.
5) Technical Specification 4.4 for surveillances of equipment required for vented fueled experiments

DISCUSSION OF R-120 LICENSE POSSESSION LIMITS CHANGE

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 limits of 35 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, neptunium, and plutonium 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

The possession limits are based on mass rather than enrichment. Any enrichment may be possessed for fueled experiments up to the mass limits requested. Experiments may use high enriched material.

The 2 grams currently allowed for foils in 2.B.(2) has been incorporated into the 35 gram mass limit of Uranium requested for License Change in 2.B.(2).

Requested Change to 2.B.(2):

> Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use in connection with operation of the reactor up to 25 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel; up to 20 grams of contained uranium-235 of any enrichment in the form of fission chambers; ~~up to 2 grams of contained uranium-235 of any enrichment in the form of foils~~*, up to 35 grams of uranium-235 of any enrichment, 1 gram of neptunium-237, and 5 grams of plutonium-239 for fueled experiments*; up to 200 grams of plutonium-239 in the form of plutonium-beryllium neutron sources; and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility.

DISCUSSION OF TECHNICAL SPECIFICATION CHANGES

TS 1.3.10.f – Fueled Experiment

The current definition of a fueled experiment was revised to clarify what is classified as a fueled experiment. The new definition exempts samples or materials containing small amounts of uranium and is discussed in detail in Section 10 and Section 14 Calculation 9.

Fueled experiments are defined for experiments involving the neutron irradiation of uranium above a specified limit for fission rate. This limit was calculated for experiments containing uranium based on limiting the radiation dose from a potential release to a TEDE of 0.001 rem to members of the public in occupied areas outside the reactor building. Experiments involving the neutron irradiation of uranium below this limit for fission rate are not classified as fueled experiments. The neutron irradiation of any amount of any other fissionable material is classified as a fueled experiment. All vented experiments that are designed to release fission gases and halogens are classified as fueled experiments.

Exclusions are given for experiments that do not involve a neutron fluence. The hazard with fueled experiments is the production and release of fission products. If no fissions occur, then the hazard is not present. An example of the “fissionable material not subjected to neutron fluence” is utilization of the Positron Beam Facility spectrometers at the NCSU reactor, which have no associated neutron fluence to experimental samples and may be used to evaluate samples containing fissionable materials. Handling precautions and other controls are observed for any experiment sample.

Detectors containing fissionable material, such as fission chambers, which are used in the operation the reactor or reactor experimental facilities are not classified as fueled experiments. Sealed sources are

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. Examples of sealed sources are 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.

Technical Specification 1.3.10.f is revised to:

> f. **Fueled Experiment**: A fueled experiment is an experiment which involves any of the following:
>
> i. Neutron irradiation of uranium exceeding 2.0×10^{6} fissions per second.
>
> ii. Neutron irradiation of any amount of other fissionable material.
>
> iii. A planned release of fission gases or halogens.
>
> Fueled Experiments exclude:
>
> iv. Fissionable material not subjected to neutron fluence.
>
> v. 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 (i.e., 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) with registration certificates, special form radioactive material as defined in 10 CFR Part 71, and PULSTAR reactor fuel elements in cladding.

TS 3.5 – Radiation Monitoring Equipment

TS 3.5 was changed to include monitoring of the vented fueled experiment exhaust for radioactivity and flow rate.

Two instruments for vented fueled experiments were added to TS 3.5:

- The vented fueled experiment exhaust gas radiation monitor is added as a requirement to verify compliance with TS 3.8.d.iii.(4). Fission gas (Kr and Xe) activity released from a vented fueled experiment provides an immediate assessment of compliance with TS 3.8 due to the mobility of noble gases. Also, radioisotopes of Kr and Xe represent the major contribution to public dose from the expected release as analyzed and therefore are the major concern.
- The vented fueled experiment exhaust gas flow rate monitor is added as a requirement to verify compliance with TS 3.8.d.iii.(3).

Alarm setpoints for the reactor stack radiation monitors were changed based on not exceeding a potential public TEDE of 0.003 rem from a fueled experiment accident. The stack gas monitor alert setpoint is set for abnormal releases that exceed those permitted for vented fueled experiments. The alert setpoint is lower than the alarm setpoint. The vented fueled experiment exhaust radiation monitor is set to isolate

the experiment exhaust and initiate the confinement system if the release exceeds TS limits. The vented fueled experiment exhaust gas flow rate monitor annunciates if the gas flow rate exceeds the setpoint specified for each experiment. These setpoints were chosen to avoid the unintended initiation of the confinement system from a planned and analyzed vented fueled experiment while ensuring the TS radiation dose limit is met.

Details on the new setpoints are provided in Calculation 10. The bases of TS 3.5 was changed to include a brief discussion of the vented fueled experiment monitors and revised setpoints.

Detection of releases below the limits for notification of unusual event Emergency Action Level (EAL) continues to be provided. Exceeding the alarm set point initiates the confinement system and isolates the exhaust from a fueled experiment accident or an abnormal release. Due to detector response and use of the confinement system the radiation dose to the public would be less than 0.003 rem from a fueled experiment release.

Technical Specification 3.5 is revised to include:

d. Vented fueled experiment exhaust gas monitor continuously monitors the experiment exhaust gas.[7][8]

e. Vented fueled experiment flow rate monitor.[7][8]

[7]Monitors for vented fueled experiments are only required while the experiment is in operation.

[8]Vented fueled experiment exhaust radiation monitor setpoint meets Specification 3.8.d.ii.

TS 3.8 – Operations with Fueled Experiments

TS 3.8 requires monitoring of the vented fueled experiment exhaust for radioactivity and flow rate when a vented fueled experiment is being performed. TS 3.5 and TS 4.4 are revised to meet TS 3.8 monitoring and surveillance requirements for vented fueled experiments.

TS 3.8 provides limiting conditions for operation for fueled experiments and is revised to establish upper limit based on three percent (3%) of the annual radiation dose limit given in 10 CFR Part 20 for members of the public outside of the reactor building 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.

   Or,

2) An accidental release from an encapsulation failure which results an instantaneous release 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 this 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 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. 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.

Radiation doses are controlled by the following:

- Minimum requirements for containers or encapsulation.
- Radiation monitoring of the stack effluent and vented experiment exhaust gas.
- Filtration and delay of the vented experiment exhaust gas.
- Conservative dose calculations to occupants inside and outside the reactor building.

Requirements of TS 3.8.a through TS 3.8.f are explained in the bases as follows:

1) Specification 3.8.a requires all specifications pertaining to experiment reactivity given in TS 3.2 be satisfied thus ensuring that reactivity control of the reactor will be maintained.
2) Specification 3.8.b requires specifications TS 3.5 and TS 3.6 pertaining to the radiation monitoring and ventilation system be satisfied thus ensuring that a public dose of 0.003 rem will not be exceeded should an accidental release occur during irradiation and/or handling of a fueled experiment.
3) Specification 3.8.c requires all specifications pertaining to limitations on experiments given in TS 3.7 be satisfied thus ensuring that fueled experiments also meet the requirements for all experiments.
4) TS 3.8.d provides limitations for fissionable materials used in fueled experiments.
    a. TS 3.8.d.i lists the physical forms allowed in fueled experiments.
    b. 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. Meeting Specification 3.8.d.ii gives a TEDE less than 1 rem and the Total Organ Dose-Equivalent to the thyroid (TODE) less than 10 rem to occupants inside the reactor building.
    c. Specification 3.8.d.iii 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 for members of the public given in 10 CFR Part 20. Radiation doses were calculated as described in *Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor*. A footnote to TS 3.8.d.iii.2 was added to specify the required filter removal efficiency.
5) Specification 3.8.e requires all specification pertaining to criticality control given in TS 5.3 be satisfied thus ensuring that fueled experiments are stored in sub-critical configurations.
6) Specification 3.8.f requires specification TS 6.2.3 and 6.5 pertaining to the review and approval of experiments be satisfied thus ensuring that fueled experiments are reviewed, approved, and documented as required.

Technical Specification 3.8 is revised in its entirety to:

3.8 Operations with Fueled Experiments

Applicability

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

Objective

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

Specifications

Fueled experiments may be performed in experimental facilities of the reactor with 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, with the exception that containers used for vented fueled experiment shall meet specification 3.8.d.iviii.1.

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

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

    ii. Any mixture of fissionable material is permitted provided that the radiation dose to members of the public outside the reactor building is less than three mrem[(1)] for members of the public Part .

    iii. Vented fueled experiments shall also meet the following:

        1. Fission gases and halogens may be released.

        2. Filtration of experiment exhaust for particulates and halogens.[(2)]

        3. Monitoring of the experiment exhaust flow rate.

        4. Monitoring of the experiment exhaust gas for radioactivity with a setpoint meeting Specification d.ii.

        5. Monitoring for halogens in the stack particulate radiation monitoring channel.

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.

[(1)] Total Effective Dose-Equivalent as defined in 10 CFR Part 20 and calculated as described in the Safety Analysis Report.

[(2)] At flow rates specified for the vented fueled experiment, removal efficiency for the filter train shall be 0.99 or greater. Individual filter removal efficiency shall be 0.95 or greater.

TS 4.5 – Radiation Monitoring Equipment

TS 4.5 is revised to include annual calibration of the vented fueled experiment exhaust gas radiation monitor and the exhaust gas flow rate monitor. Certification of iodine adsorption and periodicity of adsorbent filter replacement were also added to TS 4.5.

Annual calibration is based on calibration frequency for other monitors. Annual calibration includes a channel calibration for these monitors, a test of the isolation of the vented experiment exhaust, initiation of the confinement filters, and Control Room annunciation.

Filter removal efficiency shall be certified by the supplier to be 0.95 or greater at the specified flow rates used. The replacement time of 2 years, up to 30 months, is consistent with TS 4.6 for confinement system filters and is based on a shelf life of up to 5 years and noting that the exposure and operating characteristics to the confinement filters is similar. To meet LCOs for operation and surveillances, the confinement filters are continuously available for use and operated for a few minutes every week. The reactor building air is low relative humidity and at room temperatures, which is similar to the conditions for the vented fueled experiment exhaust.

Weekly setpoint verification is not changed in TS 4.5. This verification continues and applies to all equipment listed in TS 3.5.

Technical Specification 4.5 is revised to include:

d. Vented fueled experiment exhaust gas radiation and flow monitors shall be calibrated prior to initial operation of the experiment and annually thereafter for as long as the experiment is in operation.

e. Filter replacements for vented fueled experiments shall be biennial and shall have a removal efficiency for iodine adsorption of 0.95 or greater at the specified flow rates used.

FUELED EXPERIMENT ANALYSIS

INTRODUCTION

Information is provided in this analysis that supports the Final Safety Analysis Report (FSAR) for the license renewal of the NCSU PULSTAR nuclear reactor and changes to Technical Specifications (TS) for defining and limiting fueled experiments. Radiation dose from the release of fission gases and halogens is used as TS limits for fueled experiments. The TS limits are well below regulatory limits and provide a consistent basis for fueled experiments. This approach allows fueled experiments to be performed using any fissionable material under various experimental conditions that meet TS radiation dose criteria.

Each fueled experiment is limited to a Total Effective Dose Equivalent (TEDE) of 0.003 for members of the public outside the reactor building from the release of fission gases and halogens in a calendar year. A fueled experiment may extend beyond a calendar year for another TEDE of 0.003 rem. There may be more than one fueled experiment per calendar year with a total TEDE greater than 0.003 rem. For all airborne effluent, 10 CFR Part 20 and TS reporting requirements are followed. Reports are made as required if the TEDE from airborne effluent exceeds 0.01 rem per calendar year and 0.1 rem per calendar year. Monitoring, experiment and administrative controls, and frequent dose assessment from airborne effluent is made to ensure the constraint dose of 0.01 rem per calendar year is not exceeded. TS require an experiment specific radiation dose analysis. TS radiation dose and experiment conditions and controls are prospectively analyzed to meet TS requirements. Air monitoring and sampling are made during the release as required by TS and facility procedures providing sufficient time to halt or alter the release thereby ensuring compliance with TS and 10 CFR Part 20. It is important to note that compliance with the TS radiation dose criteria is made using radiation monitor and air sample data.

In this analysis, equations are provided for calculation of the radiation dose release inside and outside the reactor building from fission gases and halogens released for an accidental encapsulation failure and for a vented experiment. Radiation dose from the release of fission gases and halogens is controlled by limiting the fission rate of the fueled experiment. Radiation doses from released radionuclides vary due to differences in the fission yields, fission cross-sections, the thermal to non-thermal fluence rate ratio, and release conditions. Radiation dose is calculated based on time-integrated exposures, exposure time, weather conditions, and dose conversion factors. Credit for filtration, decay, and dilution is taken in the radiation dose calculations. Examples of the calculations used in the analysis of fueled experiments are described and provided in detail for a wide range of different experimental parameters and conditions. Additionally, limits and controls for common mixtures of fissionable materials, radiation monitoring, emergency planning, and security are reviewed in this analysis.

Sections 1 through 13 of this analysis provide information and equations used in the calculations. Section 14 provides example calculation data and results. Section 15 discusses factors that affect calculated values. Section 16 provides conclusions used to support the requested changes to TS Limiting Conditions for Operation (3.5 and 3.8), TS Surveillance (4.4), and TS Definitions (Section 1).

Methodology for the Review of Proposed Fueled Experiments

When fueled experiments are proposed, the methodology discussed below will be applied to review experimental parameters for compliance with TS, and for determining setpoints for related instrumentation.

Each new fueled experiment and any changes to previously approved fueled experiments shall be analyzed before the experiment is conducted to determine that radiation dose from airborne release of

fission gases and halogens does not exceed a TEDE of 0.003 rem in public areas outside the reactor building. The experiment analysis shall also verify that the TEDE is less than 1 rem and thyroid TODE is less than 10 rem inside the Reactor Building. The radiation dose analysis shall be made as described in this report. Results shall be included in the experiment review documentation.

Using the analysis methods given in this report and facility procedures, the following process is used to analyze new or changed fueled experiments:

1. Verify the fissionable material(s) is (are) allowed for use at the reactor.
2. Determine the amounts of each fissionable material that is present initially and is produced during the fueled experiment; e.g. Pu-239 from irradiation of U.
3. Perform calculations as described in Section 14 of this report for one of the examples given and for the new or changed fueled experiment. Verify the example results are correct.
4. Determine experiment controls to meet radiation dose limits given in TS. These include fission rate, mass of fissionable material used, irradiation time, and radiation monitor setpoints.
5. Add the radiation dose analysis to the experiment documentation package for review.

Data and Results:

For all fueled experiments the following parameters shall be specified; fissionable material mass, cross-sections, fluence rates, thermal and non-thermal fission yields per 100 fissions, irradiation time, exposure time, atmospheric dispersion parameter (X/Q value), and stack radiation monitor setpoints.

The following additional parameters are specified for vented experiments; experiment volume, experiment exhaust flow rate, experiment delay volume, and experiment exhaust radiation monitor setpoint.

Value of parameters are adjusted as needed such that the radiation doses do not exceed 3% of annual public radiation dose limits outside the reactor building. Document the calculation results.

Additional Analysis:

Perform other analysis regarding radiation safety, e.g. handling, storage, and disposition of the sample and mitigation and control of personnel radiation dose, airborne activity release, and contamination. Perform other analysis for the proposed fueled experiment as needed; e.g. reactivity, security, emergency response, instrumentation, compliance with the reactor license and Technical Specifications.

All new and changed fueled experiments are considered untried and require approval by RSAC and RSC. Various documentation is needed for the review and approval; e.g. radiation dose analysis, compliance with fueled experiment possession limits on the reactor license, design change, procedure change(s), experiment authorization required by NCSU Radiation Safety Manual and facility radiation protection program, and review documentation required for 10 CFR Part 50.59 and 10 CFR Part 54.

1. ASSUMPTIONS

Fueled Experiments

Conditions for planned releases from vented fueled experiments:

1. Fission gases and halogens are released. Release of particulate, powder, liquid, and solid material is prevented by design of the experiment.
2. Filters are used to remove particulates and halogens with a removal efficiency of 99.97 percent for particulates with a diameter of 0.3 microns or larger and a minimum removal efficiency of 99 percent for halogens. A filter train for halogens with multiple filters may be used in series to achieve the minimum removal efficiency. Individual halogen filter removal efficiency of 0.9 is assumed while in use.
3. A continuous, controlled release from the reactor building ventilation system during the experiment irradiation time is assumed. Decay time and experiment exhaust rate are to be specified and if used, will reduce off-site radiation dose.
4. Radiation dose is assessed for release periods of 2 h, 24 h, 96 h, and greater than 96 h.
5. 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.
6. 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.
7. Radioactive noble gases and halogens are assumed to be present at the saturation activity from irradiation at the maximum fluence rate for that particular reactor experimental facility.
8. 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.

Conditions for accidental releases from encapsulated fueled experiments:

9. Radioactive materials are encapsulated until the time of failure.
10. 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.

Conditions for accidental releases from all fueled experiments (vented or encapsulated):

11. 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.
12. 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.

13. 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.

14. 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.

15. 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.

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

17. 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.

18. The minimum reactor building free air volume is assumed to be 2400 m^3 based on reported and measured data and current design features given in TS and the Final Safety Analysis Report.

19. Confinement filter removal efficiency, or retention, is 99.97 percent for particulates with a diameter of 0.3 microns or larger and has a minimum value of 90 percent for halogens.

20. 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.

21. Reactor shut-down is assumed during an accidental release.

Fueled Experiment Definition

Conditions for fueled experiment definition:

22. 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.

23. Exclusions to fueled experiments are based on encapsulation which prevents leakage or escape of the fissionable material and fission products from intended use of the source.

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:

$$A(\infty) = k\sigma\emptyset NY \qquad \text{EQ. 2-1}$$

where, A(∞) is the saturation activity from thermal and non-thermal fission

k is a group conversion constant to give activity

k = $(1\times10^{-24}$ cm^2/barn)(1 decay/atom)(1Ci /3.7×10^{10} dps) = 2.703×10^{-23} for Ci

or k = 2.703×10^{-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 = (Mass in grams, M)(6.022×10^{23} atoms/mole)(1 mole / atomic mass number, A)

Y is the cumulative fission yield for a given radionuclide

The fission rate, or production rate, for a given radionuclide is given by the product σϕNY. Cumulative fission yields and cross-sections are constant. Saturation activity for a given fissionable material is directly proportional to the mass, fluence rate, and the ratio of thermal to non-thermal fluence rates.

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 beamtubes. 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 beamtube experiment were used in this analysis to determine the saturation activity of the fission gases and halogens that are assumed to be released. At 2 MW, the fluence rates would be twice those at 1 MW and the time to reach saturation is halved.

The following fluence rates were used in examples given in Section 14 of this analysis:

- 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)

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 from encapsulated or vented fueled experiments:

- 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. After 24 h the reactor building has experienced over 10 air changes leaving a negligible fraction of the initial concentration, (e^{-10}). A public exposure time of 24 hours is also associated with meeting emergency action levels (EAL) given in the facility emergency plan, which would not be exceeded.

For planned vented releases, exposure times used for the public were 2, 24, 96, and greater than 96 hours. 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.

4. AIRBORNE ACTIVITY CONTROLS AND FILTRATION [Ref 4, 11]

Airborne activity is controlled by sample controls, isolation of the exhaust from vented releases, radiation monitoring, and filtration.

Sample Controls

Samples from fueled experiments that are not vented are controlled by encapsulation that meets TS 3.7.

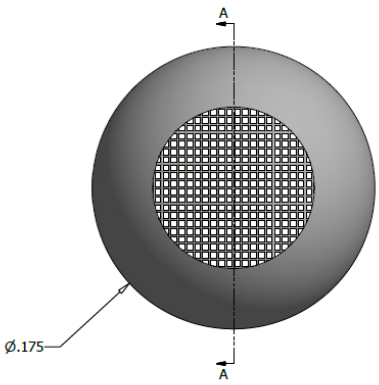
Samples from vented fueled experiments are contained to prevent the release of sample material and allow the release fission gases and halogen vapors. Samples for vented experiments are contained using a sample holder, which is placed in the experimental facility. Examples of sample holders as shown in Figure 4-1 may use a capsule or tubing with a restricted orifice (ring), mesh screen covering with smaller dimensions than the solid material, or filters to contain the sample.

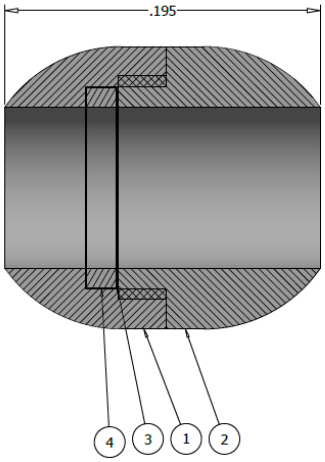
Isolation of Vented Experiment Release

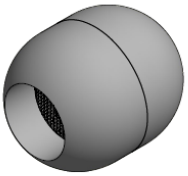
If high, unexpected activity is present in the vented fueled experiment exhaust, then the release is automatically isolated by exceeding the set point of the vented fueled experiment exhaust radiation monitor. In addition, the flow rate of the vented fueled experiment is monitored and is manually isolated if the flow rate is too high by authorized personnel.

Also, the vented experiment is halted if air sampling indicates an abnormal release is occurring or has occurred following facility health physics procedures.

Figure 4-1 – Example of a Vented Fueled Experiment Sample Holder



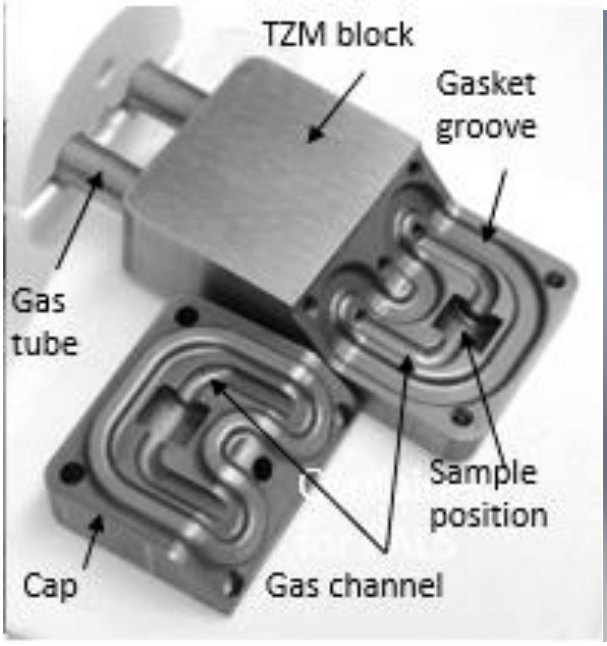


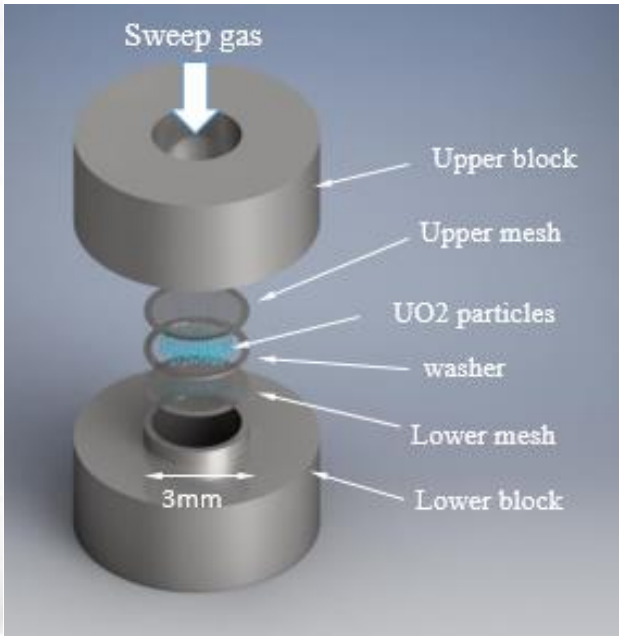


Sample Capsule with Mesh Filter

SECTION A-A
SCALE 38 : 1

PARTS LIST			
ITEM	QTY	PART NUMBER	DESCRIPTION
1	1	Sample Capsule	TZM Molybdenum
2	1	Sample Capsule Male	TZM Molybdenum
3	2	mesh1	Molybdenum
4	1	washer 1	TZM Molybdenum





Confinement System Initiation

If an alarm setpoint of the airborne activity monitors for the reactor stack is reached, then the confinement system is initiated. The setpoints for these airborne activity monitors are based on the released activity from a fueled experiment accident. High Efficiency Particulate Absorbers (HEPA) and charcoal beds are used in the confinement mode of ventilation.

Confinement filter testing

Testing is performed per TS 4.6 on the ventilation system, including filter testing in accordance with TS 4.6.d every 2 years but not to exceed 30 months. 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 for 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 fueled experiment filters and removal efficiency

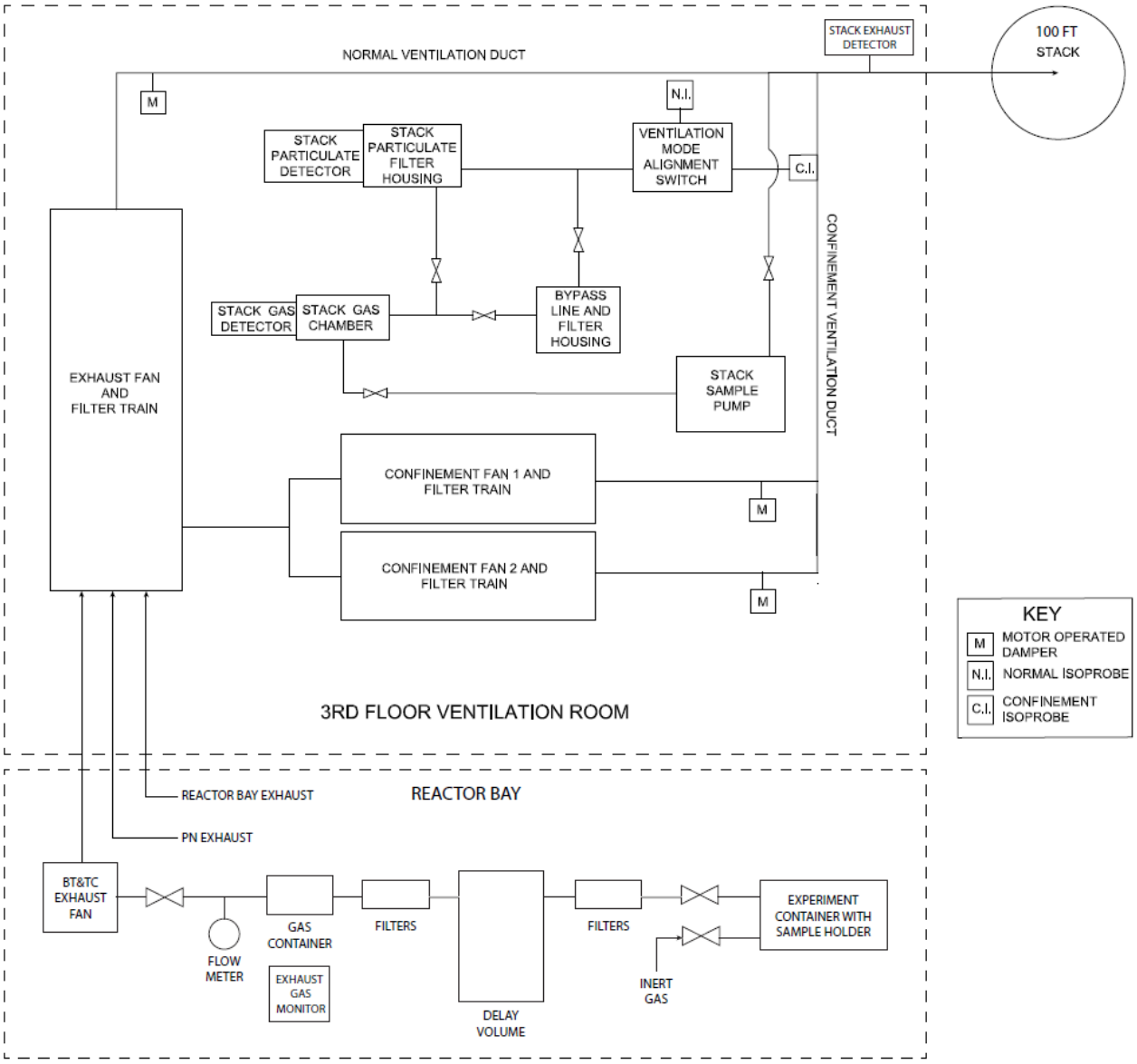
The exhaust from a vented fueled experiment is filtered for halogens and particulates prior to entering the reactor building ventilation system. A generic experiment arrangement is shown in Figure 4-2.

Commercially available HEPA filters for removal of particulates and carbon adsorber beds for iodine removal and delay of noble gases are available; e.g. activated carbon and coconut shell carbon. HEPA and carbon filters for removal of particulates and halogens are located in the experiment exhaust prior to the delay volume. A sealed housing is used to contain the HEPA and carbon filters. HEPA filter and carbon adsorber shall have a tested retention of 0.95 or greater at low flow rates. Filter certification will be reviewed and retained. Carbon adsorbers for iodine removal are tested by the vendor following an applicable standard; e.g. ASTM D 3803 "Standard Test Method for Nuclear-Grade Activated Carbon". Sufficient filtration is used to meet the required removal efficiency of 99.97% for particulates and 99% for halogens.

Certification of iodine adsorption by the supplier and adsorbent filter replacement has been added to TS 4.5 for the vented fueled experiment. The replacement time of 2 years, up to 30 months, is consistent with TS 4.6 and based on a shelf life of up to 5 years and noting that the exposure and operating characteristics to the confinement filters is similar. To meet LCOs for operation and surveillances, the confinement filters are continuously available for use and operated for a few minutes every week. The relative humidity and temperature of the vented experiment exhaust is similar to that for occupied areas in the reactor building.

Air sample filters (not shown in Figure 4-2) are used in-line after the particulate and halogen filters and before the connection to the reactor building ventilation system. Air sampling halogen filters are tested by the supplier following ASTM D 3803 with a reported retention factor greater than 95 percent for methyl iodine at low flow rates. Particulate air sampling filters shall have a reported retention of 95 percent or higher. The experiment air sampling filter housing will be sealed. Air sampling of potential releases and airborne effluent is required by the facility Radiation Protection Program. Air sampling of experiments with potential airborne release is performed periodically, typically weekly, and may be performed continuously. If the analysis of these air samples indicates an abnormal release, then the experiment is stopped and new exhaust filters are installed.

Figure 4-2 – Generic Diagram of a Vented Fueled Experiment



Description of generic vented fueled experiment components:

- A pressurized inert gas is controlled by a regulator to provide a specified flow rate.
- The experiment container allows only fission gases and halogens to be released into the experiment exhaust.
- The delay volume may consist of a long length of coiling tubing, several small tanks in series, a charcoal bed for delaying noble gas release, or a well-mixed large tank.
- Filters include particulate and charcoal adsorbers in a sealed housing located before and after the delay volume. Filters may be used in series to obtain the required removal efficiency. Redundant filter trains may be used to allow for decay prior to change out or maintenance and to provide a full capacity back up set of filters should a filter fail.

- Flow meter with a set point at the specified experiment exhaust flow rate. TS 3.5 is revised to include the vented fueled experiment flow rate monitor.
- Fission gases are monitored. The reactor stack and experiment exhaust radiation monitor setpoints are determined as described in Section 14 Calculation 10 of this analysis to meet the TS radiation dose limit. TS 3.5 is revised to include the vented fueled experiment exhaust monitor.
- Gas container with a known volume will be used with the experiment exhaust radiation monitor.
- Experiment exhaust is connected to the reactor building ventilation system upstream of the confinement fan duct.
- Isolation valves are located at the inert gas exit (experiment entry) and experiment container exit (experiment exhaust exit prior to connection to the reactor building ventilation system).

Due to the possibility of an activity release above TS limits being present in airborne effluent, isolation of the vented fueled experiment exhaust and initiation of the confinement system occurs if the vented fueled experiment exhaust radiation monitor exceeds the alarm setpoint. This monitor provides early detection and control of airborne releases since the fission gases are at higher source concentrations and not diluted. If this radiation monitor fails, the confinement system is initiated.

Due to the possibility of an abnormal release of radioactive gases or radioiodine being present in airborne effluent, initiation of the confinement system occurs. For fueled experiment releases, initiation of the confinement system occurs as follows:

- Stack gas radiation monitor or stack exhaust radiation monitor exceeding an alarm set point. Alarm setpoints are based on the maximum concentration of fission gases and ^{41}Ar released during normal operations. The alarm setpoint is set below the initial concentrations of fission gases released from a fueled experiment accident.
- Stack particulate radiation monitoring channel exceeding an alarm set point. The stack particulate monitor is equipped with a particulate filter to detect decay products of fission gases (Rb-88, Rb-89, Cs-138) and a radioiodine cartridge (e.g. TEDA charcoal or silver zeolite) to detect halogens (I-131, I-132, I-133, I-134, I-135). The alarm setpoint is based on halogen releases from a vented fueled at the TS limit.
- If the activity released from a vented fueled experiment is high and the vented fueled experiment exhaust radiation monitor fails or a filter fails, then the radiation monitors sampling the stack exhaust (stack gas or stack particulate or stack exhaust) provide control room annunciation. Operator action, as deemed necessary, is needed to initiate the confinement system. Continued operation with a radiation monitor annunciation is not a normal condition so is immediately investigated. If the activity from a vented fueled experiment release exceeds the alarm setpoint for the stack gas or stack exhaust or stack particulate radiation monitors, the confinement system is initiated.
- If any of the stack radiation monitors fail, the confinement system is initiated.
- Upon initiation of the confinement system, the normal fans stop and one of the confinement fans starts. If the first confinement fan fails to start, the second confinement fan will start.

Leakage from the vented fueled experiment filters is unlikely due to low flow rates and sealing of the filter housings and connections. Leak testing is performed using facility procedures after new filters are installed and prior to initial use; e.g. by pressure testing. Replacement of the vented fueled experiment

filters is performed using facility procedures. Should the experiment filters fail, radiation monitors would detect abnormal radioactivity, which then would isolate the experiment exhaust and initiate the confinement system. Additionally, post filter in-line air sample analysis is used to determine experiment filter performance and area air sampling is performed to determine if there is a release to the reactor building using facility procedures.

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 2.4×10^3 m^3. Additional equipment, modifications, or experiments in the reactor building significantly affecting free air volume are not expected. Therefore, the value of 2.4×10^3 m^3 was used in this analysis.

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:

$$C(0) = A(\infty)/V \qquad \text{EQ. 5-1}$$

where, C(0) in Ci/m^3 and A(∞) 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:

$$\langle C\rangle = \int C(0)e^{-kt}dt = C(0)\left[\left(1-e^{-kT}\right)/(kT)\right] \qquad \text{EQ. 5-2}$$

where, $k = \lambda + v$ in h^{-1}

$v = 0.43\ h^{-1}$ at 600 cfm exhaust rate in confinement

$0.43\ h^{-1}$ = (28,317ml per cubic foot)(600 ml/ min) (60 min/h) / 2.4×10^9 ml

$v = 1.33\ h^{-1}$ at 1870 cfm in normal ventilation

$1.33\ h^{-1}$ = (28,317 ml per cubic foot)(1870 ml/ min) (60 min/h) / 2.4×10^9 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:

$$Q = C[1 - R]F \qquad \text{EQ. 5-3}$$

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 fueled experiments

A continuous, controlled release during the experiment irradiation time occurs for vented fueled 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 for each filter. Multiple halogen filters are used to obtain a total removal efficiency of 99 percent.
- Each vented fueled experiment has a specified decay time and exhaust rate. At the maximum experiment exhaust rate a minimum experiment holdup volume in needed to give the specified decay time.

  In addition, the experiment volume is to be 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 reactor building ventilation system 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 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. Experiment exhaust flow rate is indicated locally. The release is monitored for radioactivity with indication locally and in the control room. Radiation monitor alarm annunciation is provided locally and in the control room.

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(∞) is assumed to be dispersed and held in the volume of a vented experiment, w, for decay giving the decayed, unfiltered saturation concentration, c(∞):

$$c(\infty) = [A(\infty)/w]e^{(-\lambda t)} \qquad \text{EQ. 5-4}$$

where, c(∞) in Ci/m^3

A(∞) is in Ci

w is the experiment hold up volume

Decay time, t

Concentration from a vented experiment at the reactor stack

c(∞) 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:

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

substituting Eq. 5-4 into Eq. 5-5 gives,

$$C = [A(\infty)/w]e^{(-\lambda t)}[p/(p+F)] \qquad \text{EQ. 5-6}$$

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

[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 reactor building ventilation system. The filtered release rate, q, from a vented experiment entering the reactor building ventilation system is given by:

$$q = c(\infty)[1 - R]p \qquad \text{EQ. 5-7}$$

substituting Eq. 5-4 into Eq. 5-7gives,

$$q = [A(\infty)/w]e^{(-\lambda t)}[1 - R]p \; and \; q = Q \qquad \text{EQ. 5-8}$$

where, q is the decayed, filtered release rate in Ci/s that enters the reactor building ventilation system

p is the experiment exhaust rate in ml/s

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 fueled experiment radiation and flow monitoring

The vented fueled experiment exhaust gas is monitored for flow rate and radioactivity. The experiment exhaust flow rate is specified for each planned vented fueled experiment. The experiment exhaust flow rate is indicated locally. A vented fueled experiment exhaust gas radiation monitor is used for measuring radioactivity in the exhaust gas from a vented fueled experiment. The stack gas, stack exhaust, and stack particulate radiation monitors are used to assess airborne effluent. The stack particulate monitor is equipped with a radioiodine cartridge during fueled experiments as required by the revised TS 3.8. The stack particulate monitor will detect abnormal radioiodine activity being released. Kr and Xe fission gases are readily released and therefore are monitored to provide an immediate assessment of the released activity. Radiation monitor readings are indicated locally and in the control room. Radiation monitor alarm annunciation is provided locally and in the control room.

Setpoints for radiation monitors are determined as described in Section 14 Calculation 10 to keep public dose below a TEDE of 0.003 rem. Exceeding the setpoint of the vented experiment exhaust gas radiation monitor isolates the exhaust from a vented fueled experiment and initiates the confinement system.

Setpoints for the stack radiation monitors allow normal, expected releases from a vented fueled experiment to occur without causing an alarm signal. Abnormal releases above normal vented fueled experiment operating levels provides control room annunciation. If an accidental release from an encapsulated fueled experiment at the TS dose limit occurs, the stack radiation monitor alarm initiates the confinement system.

The radiation monitor response is immediate relative to the exposure times used for calculating radiation dose to members of the public; e.g. 2 minutes for initiating the confinement system and stopping the experiment compared to an exposure time of 24 hours used in the radiation dose calculations. Therefore, the alarm would occur well before a public radiation dose of a TEDE 0.003 rem is reached.

The revised TS 4.5 includes calibration of the vented fueled experiment exhaust radiation monitor and flow rate meter. Annual calibration is required by TS and includes a channel calibration of the vented fueled experiment radiation monitor to test isolation of the experiment exhaust, initiation of the confinement system, and Control Room annunciators.

Halogens are sampled at the experiment exhaust and reactor stack. Analysis of these samples is performed upon filter removal as required by facility procedures. Radiation dose assessment of airborne effluent is routinely made using facility procedures throughout the calendar year which permits early detection of radiation dose before reaching TS limits and compliance with regulatory dose constraint levels, and regulatory dose limits.

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

Atmospheric dispersion calculation methodology in the FSAR and this amendment use the Gaussian Plume Model (GPM) at distances from 30 m to 5000 m for all exposure times. In addition, this analysis considered fumigation (i.e. trapping) conditions caused by an inversion and calm winds for periods up to 24 hours.

Gaussian Plume Model (GPM)

The atmospheric dispersion parameter [X/Q] used in this amendment are for actual occupied locations. The FSAR used the maximum calculated atmospheric dispersion {X/Q] value at a given distance regardless of the ability for occupancy. The GPM equation was used to calculate the atmospheric dispersion parameter [X/Q] for Pasquill-Gifford (PG) weather stability classes A through F:

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

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 σ_y, the lateral dispersion parameter in m, and σ_z, 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.

Decay Corrections

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.

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 range from 32 m to 70 m depending on the wind speed and stack exhaust velocity. For simplicity in making dose estimates, the actual stack height of

30 m is used for dose estimates. After the release occurs, the effective stack height may be used to determine a more realistic atmospheric dispersion parameter [X/Q] for use in dose assessment 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:

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

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):

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

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):

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

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 equation 6-5:

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

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 (trapping during an inversion)

[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 equation 6-6:

$$[X/Q] = \frac{1}{\sqrt{2\pi}h\mu\sigma_y} \qquad \text{EQ. 6-6}$$

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

The equation for fumigation was taken from Refs 19, 33, and 34. In fumigation conditions, the vertical dispersion is uniform from ground level to the stack height.

Inversions are associated with fumigation (i.e. trapping) conditions. Inversion frequency is given in FSAR Table 2-18 as ranging from 32 to 43 percent in Greensboro, NC. Inversion duration is not given in the FSAR. Fumigation in this analysis used a period of 24 hours at a wind speed of 1 m/s [Ref 12,34].

Ground based inversions occur by rapid cooling of the ground on cloudless nights with light winds. With warming of the ground during the day by the sun, the inversion ends. It is noted that the reactor building is located in an area with significant paved surface area and that the reactor typically operates during daytime hours [Ref 12].

Inversions may also occur during periods of air stagnation. In North Carolina, approximately 15 air stagnation days per year are reported [Ref 35]. Air stagnation is defined as a mean wind speed of 4 m/s

and a period of 4 days or more. This gives an annual frequency of less than five percent (15/365) for stagnant air.

It is noted that 4 m/s gives fumigation [X/Q] at one-fourth of those calculated in the FSAR and that the duration of 4 days is 4 times of that used in this analysis. It is also noted that the average weather conditions used to calculate [X/Q] are in Table 6-2 and in FSAR Table 11-10 for periods up to 4 days at wind speeds less than 4 m/s.

Maximum fumigation [X/Q] values are used to assess potential radiation dose in occupied locations near the reactor facility.

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. Ref 20 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.

For the plume centerline (y = 0 m), [X/Q] for calm winds was calculated using equation 6-7:

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

A review of weather patterns is given in Section 2 of the FSAR. Wind speed data for Jordan Hall at a height of 30 m on the university campus indicates that light winds (from 0 to2 m/s) occur approximately three percent of the time. Periods of calm winds (0 to 0.5 m/s) for Jordan Hall would occur less frequently.

Results for releases up to 24 hours

From the above discussion, periods of calm winds and fumigation are considered to be infrequent and to exist for periods up to 24 h,

In this amendment [X/Q] was calculated for calm winds at a wind speed of 0.5 m/s, fumigation at a wind speed of 1 m/s, and the GPM at a wind speed of 1 m/s for a period of 24 hours. Also, emergency action levels are associated with a period of 24 hours regarding airborne radioactive effluent.

[X/Q] Dose calculations made for calm winds and fumigation for actual occupied locations are more conservative than those made using the GPM.

From Table 14-10 for periods of 24 hours or less, the maximum [X/Q] values for occupied locations for:

- Fumigation exceeded those for the GPM for all distances
- Calm winds exceeded those for the GPM for distances from 30 m to 50 m

Atmospheric dispersion for actual occupied locations under calm winds and fumigation conditions as described in this analysis will be added to Section 11 of the FSAR for license renewal.

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), μ	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
1 to 4 days	F	0.6	1	2	Sector Averaged
> 4 days	C	0.333	0.15	3	0, centerline
> 4 days	D	0.333	0.15	2	Sector Averaged
> 4 days	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:

- 8.54×10^{-3} s/m^3 for a release times of 2 hours and 24 hours
- 7.79×10^{-4} s/m^3 for a release time from 24 to 96 hours
- 9.15×10^{-5} s/m^3 for a release time greater than 96 hours

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

[X/Q] using GPM equations given in Section 11 of the FSAR and this analysis for occupied locations give the same values. For example:

- Table 14-10 and FSAR Figure 11-2 for (x,z) location (200m, 10m vs 12m);

  Talley and Reynolds for 2-24 hours: [X/Q] = 1.93×10^{-4} s/m^3
- Table 14-10 and FSAR Figure 11-3 for (x,z) location (350m, 20m);

  North for 96 hours: [X/Q] = 6.00×10^{-5} s/m^3
- Table 14-10 and FSAR Figure 11-5 for (x,z) location (150m, 30m);

  DH Hill for greater than 96 hours: [X/Q] = 9.15×10^{-5} s/m^3

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:

$$\Psi_r = \langle C \rangle T \qquad \text{EQ. 7-1}$$

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:

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

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

Alternately,

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: 1×10^{-6} Ci/µCi and 1×10^{6} ml/m^3 gives 1 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 greater than 96 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 with 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:

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

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 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 an accidental release from any fueled experiment is limited to the following radiation doses:

- TEDE to occupants inside the reactor building is limited to 1 rem.
- Thyroid TODE to occupants inside the reactor building is limited to 10 rem.
- TEDE for members of the public is limited to 0.003 rem.

Dose from a vented fueled experiment is limited to a TEDE of 0.003 rem for 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:

$$Stack\ Activity = A(stack) = 3.93C \qquad \text{EQ. 8-1}$$

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:

$$Overhead\ Plume\ Activity = A(plume) = 56.6C \qquad \text{EQ. 8-2}$$

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 beamtube 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:

$$A(d) = 4q \qquad \text{EQ. 8-3}$$

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:

$$D = \Psi DCFf \qquad \text{EQ. 8-4}$$

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 in 2000 h as stated 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:

$$f = f'\mathrm{Gk} = \mu_{en}\mathrm{RGk} \qquad \text{EQ. 8-5}$$

f = $(4.92\times10^{-5}$ / cm)(905 cm)(2)(1.1) = 9.8×10^{-2} ~ 0.1

Alternately, *f* = 2k[1-exp(-μ_{en} r)] = 2(1.1)[1- exp(-4.92×10^{-5}*905)] ~ 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 μ_{en} *r* .

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

r = effective radius of 905 cm based on the reactor building volume of 3.0×10^{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.

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

Radiation Dose

An experiment specific 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 that meet a TEDE of 0.003 rem to occupied areas outside the reactor building and 1 rem TEDE and 10 rem TODE to the thyroid inside the reactor building.

Each experiment specifies the type, form, and amount of fissionable material(s) used, irradiation time, neutron fluence rate, experimental facility used, and exhaust rate and decay time if it is vented. Then, the experiment specific fission rate is determined that meets TS radiation dose limits is calculated using equations given in this analysis.

Examples of fission rate limits for fueled experiments and vented fueled experiments are given in Section 14 of this analysis for U-235 and Pu-239. Results for other fissionable materials are also provided in Section 14 of this analysis for example experiment conditions.

For mixtures, the most restrictive fissionable material present is the one that produces most of the public radiation dose and is used to determine the fission rate limit for that particular mixture.

Releases are monitored and assessed during the irradiation to assess radiation dose using facility procedures and to determine compliance with TS and regulatory limits.

The activity release rate is related to the fission rate. The fission rate limit is given by:

$$[f/s]^{limit} = \left(3 \times 10^{-3} rem/_{public\ TEDE\ rem\ per\ unit\ fission\ rate}\right) \qquad \text{EQ. 9-1}$$

where, 3×10^{-3} rem TEDE is the radiation dose criterion for members of the public outside the reactor building

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 fission rate. From the fission rate, the number of atoms is calculated for the fluence rate used in the fueled experiment:

$$N = [f/s]^{limit}/[\sigma\phi] \qquad \text{EQ. 9-2}$$

where, N is converted to mass of the fissionable material

The fission rate limit is based on the radiation dose criteria 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.

Accumulation of Long-lived Radioactive Material

Production of radionuclides listed in Category 2 Quantities of Concern in 10 CFR Part 37 was analyzed for security purposes.

Sr-90, Cs-137, and Pm-147 are produced in fueled experiments. Other fission products listed in 10 CFR Part 37 are produced in insignificant quantities due to low cumulative fission yields. Eq. 2-1 is modified using the fraction of equilibrium reached during the irradiation time (T) in calculating activities of Sr-90, Cs-137, and Pm-147:

$$A(T) = k\sigma\phi NY\left[1 - e^{\lambda T}\right] \qquad \text{EQ. 9-3}$$

Irradiation time of 10 years at the fission rate limits based on a TEDE of 0.003 rem to members of the public outside the reactor building were analyzed at peak fluence rates for production of Sr-90, Cs-137, and Pm-147. Realistic irradiation times, T, are less than 2000 h, i.e. 1 work year. At 2 years, the fraction of 10 CFR Part 37 Category 2 activity limits was less than 4×10^{-4}. At 2000 h, the fraction was less than 5×10^{-6}. Therefore, at the radiation dose of 0.003 rem TEDE to the public outside the reactor building, 10 CFR Part 37 limits are not reached. Details are given in Section 14 Calculation 6.

Other Experiment Controls

TS limitations and conditions for experiments, reactivity, storage of fissionable materials and experiment reviews apply to fueled experiments. Facility procedures are used as applicable for control of all experiments. Reviews of new experiments and experiment changes are performed as stated in TS.

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 2.0×10^6 fissions per second
- Neutron irradiation of any amount of other fissionable materials
- A planned release of fission gases or halogens.

At the fission rate limit of 2.0×10^6 f/s for U, the sum of the saturation activities for Sr-90, Cs-137, and Pm-147 are less than 7 μCi.

The definition of a fueled experiment is revised. Fueled experiments involve neutron irradiation of materials containing uranium with a release of fission gases and halogens that exceed one percent (1%) of the annual public dose limit given in 10 CFR Part 20, i.e. a Total Effective Dose-Equivalent (TEDE) of 0.001 rem inside or outside the reactor building.

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.

Exclusions to fueled experiments are based on encapsulation of fissionable materials which prevent leakage or escape of the material from the intended use of the source. 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. Fissionable materials are defined in TS. 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 certificates, 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.

11. EMERGENCY PLAN [Ref 4, 6, 7]

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 as analyzed in Section 13 of the FSAR.

No revision to the emergency plan is needed.

12. SECURITY, STORAGE, and INVENTORY [Ref 11, 24, 32, Section 14 Calculation 6]

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. The fraction of 10 CFR Part 37 Category 2 activity limits is below the limiting value of 1 for extended irradiation times at the experiment fluence rates achievable and mass of fissionable materials used.

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.

13. POSSESSION LIMITS

Possession of Uranium (U) with any enrichment of U-235 up to 35 grams of U-235, up to 1 gram of Neptunium-237 (Np-237), and up to 5 grams of Plutonium (Pu) for fueled experiments is requested based on experiment needs. Experiment needs include evaluation of fissionable materials used for reactor fuel and neutron detection.

Common fissionable materials potentially present in fueled experiments include the following:

Uranium (U)

U-234, U-235 and U-238 are present in natural abundance, U-235 enriched U, or depleted U.

U-236 and U-239 are not produced in significant amounts from the activation of U. U-236 is produced by activation of U-235. Activation of U-238 produces U-239 which beta decays to Np-239 and then beta decays to Pu-239.

Neptunium (Np)

Np-237 is a long-lived radionuclide. Np-237 undergoes activation to produce Np-238 which beta decays to produce Pu-238.

Plutonium (Pu)

Pu-239 is commonly present in samples of Pu and in samples of U. Other isotopes of Pu are typically present in lower amounts. The fission reaction cross-sections for Pu-239 are significantly larger than those for Pu-238, Pu-240, and Pu-242 and lower than those for Pu-241. Pu-239 is also produced by activation of U-238 with beta decays from U-239 and Np-239.

Fissionable materials in mixtures with activation and decay products in fueled experiments include:

- U-234, U-235, U-236, U-238, Pu-239
- Np-237, Np-238, Pu-238
- Pu 238, Pu-239, Pu-240, Pu-241, Pu-242

Restrictive fissionable materials for mixtures were U-235, Np-237, and Pu-239 as determined from radiation dose per unit mass, mass, and fission rate in Section 14 Calculation 11.

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. Results for other fissionable materials are given.

- Example calculations for vented experiments were made for an experiment exhaust rate of 3 lpm (50 ml/s) and a 30 minute decay time, or decay volume of 90 liters (30 m x 3 lpm).
- The vented experiment in these example calculations is routed into the reactor building ventilation through the beamtube exhaust. The beamtube exhaust connects to the reactor building ventilation system in the ventilation room inside the reactor building.
- For accidental releases, 24 hours gave the higher time integrated exposures and radiation doses. This is due to the experiment being stopped at the time of the accident and assuming all of the released activity is ventilated from the reactor building within 24 hours. The radiation dose of 0.003 rem to areas outside the reactor building was limiting in all of the examples.
- For vented experiments, $A(\infty)$ is assumed to always be present. Radiation dose in public areas outside the reactor building is directly proportional to the product $[X/Q]T$, where T is the exposure time. T of 24 h was used in the following examples. For T other than 24 h, different results from those given in the example calculations are obtained. In general, $[X/Q]T$ has to be evaluated for the irradiation conditions for each fueled experiment to determine the radiation dose and fission rate limit.

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

Calculation 10 was performed to determine radiation monitor set points for the revised TS 3.5 for radiation monitoring.

Calculation 11 was performed to provide example results for additional fissionable materials and mixtures. Results giving 3% of the annual public dose limits outside the reactor building are given for several fissionable materials using the same equations, assumptions, and irradiation conditions used in the example calculations.

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

Table 14-1 and Table 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 1.0×10^{10} f/s for U-235 and 1.4×10^{10} f/s for Pu-239 were calculated for an exhaust flow rate of 3 lpm, decay time of 30 minutes (90 liters decay volume), and an exposure time of 24 h.

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

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	U-235		Target atoms, N	1.34E+19	atoms
Mass	5.21E-03	g	Thermal fission rate	7.81E+09	f/s
Mass Number, A	235	g/mol	Non-thermal fission rate	2.29E+09	f/s
Sigma thermal	585	barns	Total fission rate	1.01E+10	f/s
Sigma non-thermal	571	barns	Total fissions	8.14E+16	
X/Q	8.54E-03	s/m^3	Reactor volume	2.40E+09	ml
Thermal flux	1.00E+12	cm^2/s			
Non-thermal flux	3.00E+11	cm^2/s			
Max Irradiation time	8.64E+04	s	F normal	0.883	m^3/s
			v normal	3.68E-04	1/s
Vented experiment exhaust	3	lpm			
Vented experiment volume	90	liters			
Public exposure time	24	hours	(1-R) halogens exp exhaust	0.01	

Table 14-1 – Continued

ISOTOPIC DATA					
Nuclide	**Half-Life (sec)**	**Decay Constant (1/s)**	**Cumulative Yield %**		**Eq. 2-1 Saturation Activity (μCi)**
			Thermal Fission	**Non-Thermal Fission**	
^{83m}Kr	6.70E+03	1.04E-04	5.36E-01	5.75E-01	1.49E+03
^{85m}Kr	1.61E+04	4.30E-05	1.29E+00	1.36E+00	3.56E+03
^{85}Kr	3.39E+08	2.05E-09	2.83E-01	2.96E-01	7.80E+02
^{87}Kr	4.57E+03	1.52E-04	2.56E+00	2.54E+00	6.97E+03
^{88}Kr	1.02E+04	6.78E-05	3.55E+00	3.43E+00	9.62E+03
^{89}Kr	1.89E+02	3.67E-03	4.51E+00	3.97E+00	1.20E+04
^{131m}Xe	1.03E+06	6.74E-07	4.05E-02	3.54E-02	1.07E+02
^{133m}Xe	1.89E+05	3.66E-06	1.89E-01	1.97E-01	5.21E+02
^{133}Xe	4.53E+05	1.53E-06	6.70E+00	6.71E+00	1.83E+04
^{135m}Xe	9.18E+02	7.55E-04	1.10E+00	1.26E+00	3.10E+03
^{135}Xe	3.28E+04	2.12E-05	6.54E+00	6.58E+00	1.79E+04
^{137}Xe	2.29E+02	3.02E-03	6.13E+00	5.98E+00	1.66E+04
^{138}Xe	8.46E+02	8.19E-04	6.30E+00	6.00E+00	1.70E+04
^{131}I	6.93E+05	1.00E-06	2.89E+00	3.22E+00	8.09E+03
^{132}I	8.26E+03	8.39E-05	4.31E+00	4.66E+00	1.20E+04
^{133}I	7.49E+04	9.26E-06	6.70E+00	6.70E+00	1.83E+04
^{134}I	3.16E+03	2.20E-04	7.83E+00	7.63E+00	2.12E+04
^{135}I	2.37E+04	2.93E-05	6.28E+00	6.27E+00	1.71E+04
^{83}Br	8.64E+03	8.02E-05	5.40E-01	5.76E-01	1.49E+03
^{84}Br	1.91E+03	3.63E-04	9.67E-01	1.01E+00	2.66E+03

Table 14-1 – 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)**
^{83m}Kr	1.41E-07	1.52E-02	2.13E-09
^{85m}Kr	3.75E-07	1.10E+02	4.14E-05
^{85}Kr	8.89E-08	1.74E+00	1.55E-07
^{87}Kr	6.04E-07	5.25E+02	3.18E-04
^{88}Kr	9.69E-07	1.33E+03	1.29E-03
^{89}Kr	1.85E-09	1.20E+03	2.22E-06
^{131m}Xe	1.22E-08	5.48E+00	6.69E-08
^{133m}Xe	5.89E-08	1.99E+01	1.17E-06
^{133}Xe	2.08E-06	2.25E+01	4.66E-05
^{135m}Xe	9.08E-08	2.79E+02	2.53E-05
^{135}Xe	1.96E-06	1.73E+02	3.39E-04
^{137}Xe	8.19E-09	1.10E+02	9.01E-07
^{138}Xe	4.43E-07	7.10E+02	3.15E-04
^{131}I	9.19E-09	3.97E+04	3.65E-04
^{132}I	1.17E-08	1.95E+03	2.29E-05
^{133}I	2.05E-08	7.41E+03	1.52E-04
^{134}I	1.63E-08	1.89E+03	3.08E-05
^{135}I	1.85E-08	2.54E+03	4.70E-05
^{83}Br	1.47E-09	1.09E+02	1.60E-07
^{84}Br	1.58E-09	1.35E+03	2.14E-06
		TOTAL	**3.00E-03**

Supporting Calculations for U-235:

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

Kr-87: 6.97×10^{3} µCi = $(5.21\times10^{-3}g)(6.022\times10^{23} / 235\ g)(2.703\times10^{-29})$

x $[(585)(1\times10^{12})(2.56/100)+(571)(3\times10^{11})(2.54/100)]$ or 6.97×10^{-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: 6.07×10^{-7}µCi-h/ml = $(6.97\times10^{3}$µCi$/9\times10^{4}$ml$)(50)e^{[-(1.52E-4)(9E4)/50]}(8.54\times10^{-3}s/m^3)(1\times10^{-6}m^3/ml)(24\ h)$

I133: 2.05×10^{-8}µCi-h/ml = $(1.83\times10^{4}/9\times10^{4})(e^{[-(9.26E-6)(9E4)/50]})(50)(0.01)(8.54\times10^{-3})(1\times10^{-6})(24)$

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

TEDE is 4.66×10^{-5} rem = $(2.08\times10^{-6}$ µCi-h/ml$)(22.5$ rem per µCi-h/ml)

The public dose from 1.01×10^{10} f/s for U-235 is 3.0×10^{-3} rem which gives 2.97×10^{-13} rem per f/s.

Fission rate limit from Eq. 9-1: 1.01×10^{10} f/s = 3.0×10^{-3} rem / $(2.97\times10^{-13}$ rem per f/s)

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

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	Pu-239		Target atoms, N	1.47E+19	atoms
Mass	5.84E-03	g	Thermal fission rate	1.10E+10	f/s
Mass Number, A	239	g/mol	Non-thermal fission rate	3.88E+09	f/s
Sigma thermal	748	barns	Total fission rate	1.45E+10	f/s
Sigma non-thermal	789	barns	Total fissions	1.17E+17	
X/Q	8.54E-03	s/m^3	Reactor volume	2.40E+09	ml
Thermal flux	1.00E+12	cm^2/s			
Non-thermal flux	3.00E+11	cm^2/s			
Max Irradiation time	8.64E+04	s	F normal	0.883	m^3/s
			v normal	3.68E-04	1/s
Vented experiment exhaust	3	lpm			
Vented experiment volume	90	liters			
Public exposure time	24	hours	(1-R) halogens exp exhaust	0.01	

Table 14-2 – Continued

ISOTOPIC DATA					
Nuclide	**Half-Life (sec)**	**Decay Constant (1/s)**	**Cumulative Yield %**		**Eq. 2-1 Saturation Activity (µCi)**
			Thermal Fission	**Non-Thermal Fission**	
^{83m}Kr	6.70E+03	1.04E-04	2.97E-01	3.15E-01	1.18E+03
^{85m}Kr	1.61E+04	4.30E-05	5.63E-01	5.94E-01	2.23E+03
^{85}Kr	3.39E+08	2.05E-09	1.23E-01	1.38E-01	4.95E+02
^{87}Kr	4.57E+03	1.52E-04	9.89E-01	1.04E+00	3.92E+03
^{88}Kr	1.02E+04	6.78E-05	1.27E+00	1.29E+00	5.00E+03
^{89}Kr	1.89E+02	3.67E-03	1.45E+00	1.45E+00	5.69E+03
^{131m}Xe	1.03E+06	6.74E-07	4.24E-02	4.27E-02	1.66E+02
^{133m}Xe	1.89E+05	3.66E-06	2.31E-01	2.45E-01	9.19E+02
^{133}Xe	4.53E+05	1.53E-06	7.02E+00	6.97E+00	2.74E+04
^{135m}Xe	9.18E+02	7.55E-04	1.84E+00	2.08E+00	7.43E+03
^{135}Xe	3.28E+04	2.12E-05	7.60E+00	7.54E+00	2.97E+04
^{137}Xe	2.29E+02	3.02E-03	6.01E+00	5.58E+00	2.31E+04
^{138}Xe	8.46E+02	8.19E-04	5.17E+00	4.71E+00	1.98E+04
^{131}I	6.93E+05	1.00E-06	3.86E+00	3.88E+00	1.51E+04
^{132}I	8.26E+03	8.39E-05	5.39E+00	5.32E+00	2.10E+04
^{133}I	7.49E+04	9.26E-06	6.97E+00	6.91E+00	2.72E+04
^{134}I	3.16E+03	2.20E-04	7.41E+00	7.11E+00	2.87E+04
^{135}I	2.37E+04	2.93E-05	6.54E+00	6.08E+00	2.52E+04
^{83}Br	8.64E+03	8.02E-05	2.97E-01	3.15E-01	1.18E+03
^{84}Br	1.91E+03	3.63E-04	4.29E-01	4.63E-01	1.71E+03

Table 14-2 – 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 TEDE (rem)
^{83m}Kr	1.11E-07	1.52E-02	1.69E-09
^{85m}Kr	2.35E-07	1.10E+02	2.60E-05
^{85}Kr	5.64E-08	1.74E+00	9.81E-08
^{87}Kr	3.40E-07	5.25E+02	1.78E-04
^{88}Kr	5.04E-07	1.33E+03	6.71E-04
^{89}Kr	8.80E-10	1.20E+03	1.06E-06
^{131m}Xe	1.89E-08	5.48E+00	1.04E-07
^{133m}Xe	1.04E-07	1.99E+01	2.07E-06
^{133}Xe	3.11E-06	2.25E+01	7.00E-05
^{135m}Xe	2.17E-07	2.79E+02	6.05E-05
^{135}Xe	3.26E-06	1.73E+02	5.64E-04
^{137}Xe	1.14E-08	1.10E+02	1.25E-06
^{138}Xe	5.16E-07	7.10E+02	3.67E-04
^{131}I	1.72E-08	3.97E+04	6.83E-04
^{132}I	2.06E-08	1.95E+03	4.01E-05
^{133}I	3.05E-08	7.41E+03	2.26E-04
^{134}I	2.20E-08	1.89E+03	4.16E-05
^{135}I	2.72E-08	2.54E+03	6.90E-05
^{83}Br	1.16E-09	1.09E+02	1.26E-07
^{84}Br	1.01E-09	1.35E+03	1.37E-06
		TOTAL	**3.00E-03**

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

Table 14-3 through Table 14-6 below give the data and results for radiation dose from accidental releases from experiments using Pu-239. Table 14-7 through Table 14-9 give the data and results for radiation dose from accidental releases from experiments using U-235. For accidental releases, the public TEDE dose criteria for fueled experiments of 0.003 rem is used. Thyroid TODE is < 10 rem and TEDE inside the reactor building is < 1 rem. Fission rates are 9.92×10^{10} f/s for Pu-239 and 1.09×10^{11} f/s for U-235.

Table 14-3 – Calculation 3 – Accidental Release from Experiment using Pu-239

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	Pu-239		Target atoms, N	1.01E+20	atoms
Mass	4.00E-02	g	Thermal fission rate	7.54E+10	f/s
Mass Number, A	239	g/mol	Non-thermal fission rate	2.39E+10	f/s
Sigma thermal	748	barns	Total fission rate	9.92E+10	f/s
Sigma non-thermal	789	barns	Total fissions		
X/Q	8.54E-03	s/m^3	Reactor volume	2.40E+09	ml
Thermal flux	1.00E+12	cm^2/s	F confinement	0.283	m^3/s
Non-thermal flux	3.00E+11	cm^2/s	v confinement	1.18E-04	1/s
Max Irradiation time	1.83E+07	s	F normal	0.883	m^3/s
			v normal	3.68E-04	1/s
			Evacuation time in confine	120	s
			Evacuation time in normal	240	s
			NG reactor correction	0.1	s
Public exposure time	24	hours	(1-R) halogens confine	0.1	

Table 14-3 – Continued

ISOTOPIC DATA					
Nuclide	**Half-Life (sec)**	**Decay Constant (1/s)**	**Cumulative Yield %**		**Eq. 2-1 Saturation Activity (μCi)**
			Thermal Fission	**Non-Thermal Fission**	
^{83m}Kr	6.70E+03	1.04E-04	2.97E-01	3.15E-01	8.08E+03
^{85m}Kr	1.61E+04	4.30E-05	5.63E-01	5.94E-01	1.53E+04
^{85}Kr	3.39E+08	2.05E-09	1.23E-01	1.38E-01	3.39E+03
^{87}Kr	4.57E+03	1.52E-04	9.89E-01	1.04E+00	2.68E+04
^{88}Kr	1.02E+04	6.78E-05	1.27E+00	1.29E+00	3.42E+04
^{89}Kr	1.89E+02	3.67E-03	1.45E+00	1.45E+00	3.89E+04
^{131m}Xe	1.03E+06	6.74E-07	4.24E-02	4.27E-02	1.14E+03
^{133m}Xe	1.89E+05	3.66E-06	2.31E-01	2.45E-01	6.29E+03
^{133}Xe	4.53E+05	1.53E-06	7.02E+00	6.97E+00	1.88E+05
^{135m}Xe	9.18E+02	7.55E-04	1.84E+00	2.08E+00	5.09E+04
^{135}Xe	3.28E+04	2.12E-05	7.60E+00	7.54E+00	2.04E+05
^{137}Xe	2.29E+02	3.02E-03	6.01E+00	5.58E+00	1.58E+05
^{138}Xe	8.46E+02	8.19E-04	5.17E+00	4.71E+00	1.36E+05
^{131}I	6.93E+05	1.00E-06	3.86E+00	3.88E+00	1.04E+05
^{132}I	8.26E+03	8.39E-05	5.39E+00	5.32E+00	1.44E+05
^{133}I	7.49E+04	9.26E-06	6.97E+00	6.91E+00	1.87E+05
^{134}I	3.16E+03	2.20E-04	7.41E+00	7.11E+00	1.97E+05
^{135}I	2.37E+04	2.93E-05	6.54E+00	6.08E+00	1.72E+05
^{83}Br	8.64E+03	8.02E-05	2.97E-01	3.15E-01	8.08E+03
^{84}Br	1.91E+03	3.63E-04	4.29E-01	4.63E-01	1.17E+04

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 Ventilation		Normal Ventilation	
	Reactor (µCi-h/ml)	Public (µCi-h/ml)	Reactor (µCi-h/ml)	Public (µCi-h/ml)
^{83m}Kr	1.11E-07	1.02E-08	2.12E-07	1.60E-09
^{85m}Kr	2.10E-07	2.66E-08	4.05E-07	3.05E-09
^{85}Kr	4.68E-08	8.05E-09	9.02E-08	6.80E-10
^{87}Kr	3.67E-07	2.79E-08	7.01E-07	5.28E-09
^{88}Kr	4.70E-07	5.16E-08	9.03E-07	6.80E-09
^{89}Kr	4.35E-07	2.88E-09	6.93E-07	5.22E-09
^{131m}Xe	1.57E-08	2.69E-09	3.03E-08	2.28E-10
^{133m}Xe	8.68E-08	1.45E-08	1.67E-07	1.26E-09
^{133}Xe	2.59E-06	4.40E-07	4.99E-06	3.76E-08
^{135m}Xe	6.71E-07	1.63E-08	1.24E-06	9.34E-09
^{135}Xe	2.80E-06	4.09E-07	5.40E-06	4.07E-08
^{137}Xe	1.83E-06	1.41E-08	3.01E-06	2.27E-08
^{138}Xe	1.78E-06	4.05E-08	3.28E-06	2.47E-08
^{131}I	1.43E-06	2.44E-08	2.75E-06	2.08E-08
^{132}I	1.98E-06	2.00E-08	3.79E-06	2.86E-08
^{133}I	2.57E-06	4.10E-08	4.96E-06	3.74E-08
^{134}I	2.68E-06	1.63E-08	5.10E-06	3.84E-08
^{135}I	2.37E-06	3.28E-08	4.57E-06	3.44E-08
^{83}Br	1.11E-07	1.14E-09	2.13E-07	1.60E-09
^{84}Br	1.58E-07	6.82E-10	2.99E-07	2.25E-09

Supporting Calculations:

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

I-131: 1.04×10^{5} µCi = $(4.00\times10^{-2}$ g$)(6.022\times10^{23}/239$ g$)(2.703\times10^{-29})$

x $[(748)(1\times10^{12})(3.86/100)+(789)(3\times10^{11})(3.88/100)]$, or 1.04×10^{-1} Ci

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

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

where:

Ψr for each ventilation mode and exposure time is summed for the reactor building:

$k = \lambda + v$, $v = 3.68\times10^{-4}$ per s in normal ventilation and 1.18×10^{-4} per s in confinement

T = 240 s in normal ventilation and 120 s in confinement

Xe-133: $\Psi r = (1.88\times10^{5}$ µCi / 2.4×10^{9} ml$)(1$ h / 3600 s$)$

x $\{[1-\exp(-(3.68\times10^{-4}+1.53\times10^{-6})(240)) / (3.68\times10^{-4}+1.53\times10^{-6})]$ s

+ $[1-\exp((-1.18\times10^{-4}+1.53\times10^{-6})(120)) / (1.18\times10^{-4}+1.53\times10^{-6})]$ s$\}$

= $(2.59\times10^{-6} + 4.99\times10^{-6})$ µCi-h/ml

= 7.59×10^{-6} µCi-h/ml

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

$$\Psi_p = \langle C \rangle [1-R][X/Q]FT$$

where:

Ψp for each ventilation mode and exposure time is summed for the public:

T = 240 s in normal ventilation and 8.64×10^{4} s 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: $\Psi p = (1.88\times10^{5}\ \mu Ci / 2.4E9\ ml)(8.54\times10^{-3})(1\ h / 3600\ s)$

$\times \{[1-\exp(-(3.68\times10^{-4}+1.53\times10^{-6})(240)) / (3.68\times10^{-4}+1.53\times10^{-6})]\ s$

$+ [1-\exp((-1.18\times10^{-4}+1.53\times10^{-6})(8.64\times10^{4})) / (1.18\times10^{-4}+1.53\times10^{-6})]\ s\}$

$= (4.40\times10^{-7} + 3.76\times10^{-8})\ \mu Ci\text{-}h/ml = 4.78\times10^{-7}\ \mu Ci\text{-}h/ml$

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

	DCF			Eq. 8-4					
				Confinement Ventilation			Normal Ventilation		
Nuclide	Effective Inhal. rem per µCi-h/ml)	Thyroid Inhal. rem per µCi-h/ml)	Submers rem per µCi-h/ml)	Reactor TEDE (rem)	Reactor Thyroid Dose (rem)	Public TEDE (rem)	Reactor TEDE (rem)	Reactor Thyroid Dose (rem)	Public TEDE (rem)
^{83m}Kr			1.52E-02	1.68E-10	1.68E-10	1.55E-10	3.22E-10	3.22E-10	2.43E-11
^{85m}Kr			1.10E+02	2.32E-06	2.32E-06	2.93E-06	4.46E-06	4.46E-06	3.36E-07
^{85}Kr			1.74E+00	8.14E-09	8.14E-09	1.40E-08	1.57E-08	1.57E-08	1.18E-09
^{87}Kr			5.25E+02	1.93E-05	1.93E-05	1.46E-05	3.68E-05	3.68E-05	2.78E-06
^{88}Kr			1.33E+03	6.26E-05	6.26E-05	6.87E-05	1.20E-04	1.20E-04	9.06E-06
^{89}Kr			1.20E+03	5.22E-05	5.22E-05	3.46E-06	8.32E-05	8.32E-05	6.27E-06
^{131m}Xe			5.48E+00	8.61E-09	8.61E-09	1.47E-08	1.66E-08	1.66E-08	1.25E-09
^{133m}Xe			1.99E+01	1.73E-07	1.73E-07	2.88E-07	3.33E-07	3.33E-07	2.51E-08
^{133}Xe			2.25E+01	5.82E-06	5.82E-06	9.88E-06	1.12E-05	1.12E-05	8.45E-07
^{135m}Xe			2.79E+02	1.87E-05	1.87E-05	4.54E-06	3.45E-05	3.45E-05	2.60E-06
^{135}Xe			1.73E+02	4.85E-05	4.85E-05	7.09E-05	9.35E-05	9.35E-05	7.05E-06
^{137}Xe			1.10E+02	2.02E-05	2.02E-05	1.55E-06	3.31E-05	3.31E-05	2.50E-06
^{138}Xe			7.10E+02	1.27E-04	1.27E-04	2.88E-05	2.33E-04	2.33E-04	1.76E-05
^{131}I	3.95E+04	1.30E+06	2.42E+02	5.64E-02	1.85E+00	9.68E-04	1.09E-01	3.57E+00	8.24E-04
^{132}I	4.57E+02	7.73E+03	1.49E+03	1.20E-03	1.56E-02	3.89E-05	2.30E-03	2.99E-02	5.57E-05
^{133}I	7.02E+03	2.16E+05	3.92E+02	1.81E-02	5.55E-01	3.04E-04	3.50E-02	1.07E+00	2.77E-04
^{134}I	1.58E+02	1.28E+03	1.73E+03	8.86E-04	3.89E-03	3.08E-05	1.69E-03	7.40E-03	7.26E-05
^{135}I	1.47E+03	3.76E+04	1.06E+03	3.75E-03	8.94E-02	8.31E-05	7.22E-03	1.72E-01	8.74E-05
^{83}Br	1.03E+02		5.09E+00	1.15E-05	5.64E-08	1.24E-07	2.21E-05	1.08E-07	1.74E-07
^{84}Br	1.01E+02		1.25E+03	3.58E-05	1.98E-05	9.23E-07	6.76E-05	3.74E-05	3.05E-06
			TOTAL	**8.08E-02**	**2.52E+00**	**1.63E-03**	**1.56E-01**	**4.85E+00**	**1.37E-03**

Supporting Calculations:

Dose calculations (Reference Eq. 8-4): $D = \Psi \cdot DCF \cdot f$

Xe-133 dose inside the reactor building:

$$1.70\times10^{-5} \text{ rem} = [(2.59\times10^{-6} + 4.99\times10^{-6}) \mu\text{Ci-h/ml}](22.5 \text{ rem per } \mu\text{Ci-h/ml})(0.1)$$

$$= (5.82\times10^{-6} + 1.22\times10^{-5}) \text{ rem}$$

Xe-133 dose outside the reactor building:

$$1.07\times10^{-5} \text{ rem} = [(4.40\times10^{-7} + 3.78\times10^{-8}) \mu\text{Ci-h/ml}]((22.5\text{rem per } \mu\text{Ci-h/ml})$$

$$= (9.88\times10^{-6} + 8.45\times10^{-7}) \text{ rem}$$

I-131 dose inside the reactor building:

$$\text{TEDE is } 1.65\times10^{-1} \text{ rem} = (1.43\times10^{-6} + 2.75\times10^{-6})(3.95\times10^{4}+0.1(242))$$

$$= (5.64\times10^{-2} + 1.09\times10^{-1}) \text{ rem}$$

$$\text{Thyroid TODE is } 5.42 \text{ rem} = (1.43\times10^{-6} + 2.75\times10^{-6})(1.3\times10^{6}+0.1(242))$$

$$= (1.85 + 3.57) \text{ rem}$$

I-131 dose outside the reactor building:

$$\text{TEDE is } 1.79\times10^{-3} \text{ rem} = (2.44\times10^{-8} + 2.08\times10^{-8})(3.95\times10^{4} + 242)$$

$$= (9.64\times10^{-4} + 8.24\times10^{-4}) \text{ rem}$$

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

Parameter	Reactor Building		Public Areas
	TEDE (rem)	Thyroid TODE (rem)	TEDE (rem)
Total dose in rem	2.36E-01	7.37E+00	3.00E-03
rem per f/s	2.38E-12	7.42E-11	3.02E-14
Dose limit in rem	1.00E+00	1.00E+01	3.00E-03
Fission rate limit (f/s): Eq. 9-1	4.20E+11	1.35E+11	9.92E+10

Supporting Calculations:

Fission rate (Reference Eq. 9-1):

The TEDE < 1 rem and Thyroid TODE < 10 rem inside the reactor building. Therefore, the fission rate limit for an accidental release from Pu-239 irradiation is based on public dose outside the reactor building:

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

$$9.92\times10^{10} \text{ f/s} = (0.003 \text{ rem}) / (3.02\times10^{-14} \text{ rem per f/s})$$

Table 14-7 – Calculation 4 – Accidental Release from Experiment using U-235

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	U-235		Target atoms, N	1.44E+20	atoms
Mass	5.61E-02	g	Thermal fission rate	8.41E+10	f/s
Mass Number, A	235	g/mol	Non-thermal fission rate	2.46E+10	f/s
Sigma thermal	585	barns	Total fission rate	1.09E+11	f/s
Sigma non-thermal	571	barns	Total fissions		
X/Q	8.54E-03	s/m^3	Reactor volume	2.40E+09	ml
Thermal flux	1.00E+12	cm^2/s	F confinement	0.283	m^3/s
Non-thermal flux	3.00E+11	cm^2/s	v confinement	1.18E-04	1/s
Max Irradiation time	1.40E+07	s	F normal	0.883	m^3/s
			v normal	3.68E-04	1/s
			Evacuation time in confine	120	s
			NG reactor correction	0.1	s
Public exposure time	24	hours	(1-R) halogens confine	0.1	

Table 14-7 – Continued

ISOTOPIC DATA

Nuclide	Half-Life (sec)	Decay Constant (1/s)	Cumulative Yield %		Eq. 2-1 Saturation Activity (µCi)
			Thermal Fission	Non-Thermal Fission	
^{83m}Kr	6.70E+03	1.04E-04	5.36E-01	5.75E-01	1.60E+04
^{85m}Kr	1.61E+04	4.30E-05	1.29E+00	1.36E+00	3.84E+04
^{85}Kr	3.39E+08	2.05E-09	2.83E-01	2.96E-01	8.40E+03
^{87}Kr	4.57E+03	1.52E-04	2.56E+00	2.54E+00	7.51E+04
^{88}Kr	1.02E+04	6.78E-05	3.55E+00	3.43E+00	1.04E+05
^{89}Kr	1.89E+02	3.67E-03	4.51E+00	3.97E+00	1.29E+05
^{131m}Xe	1.03E+06	6.74E-07	4.05E-02	3.54E-02	1.16E+03
^{133m}Xe	1.89E+05	3.66E-06	1.89E-01	1.97E-01	5.61E+03
^{133}Xe	4.53E+05	1.53E-06	6.70E+00	6.71E+00	1.97E+05
^{135m}Xe	9.18E+02	7.55E-04	1.10E+00	1.26E+00	3.34E+04
^{135}Xe	3.28E+04	2.12E-05	6.54E+00	6.58E+00	1.92E+05
^{137}Xe	2.29E+02	8.19E-04	6.30E+00	6.00E+00	1.83E+05
^{138}Xe	8.46E+02	1.00E-06	2.89E+00	3.22E+00	8.71E+04
^{131}I	6.93E+05	8.39E-05	4.31E+00	4.66E+00	1.29E+05
^{132}I	8.26E+03	9.26E-06	6.70E+00	6.70E+00	1.97E+05
^{133}I	7.49E+04	2.20E-04	7.83E+00	7.63E+00	2.29E+05
^{134}I	3.16E+03	2.93E-05	6.28E+00	6.27E+00	1.84E+05
^{135}I	2.37E+04	8.02E-05	5.40E-01	5.76E-01	1.61E+04
^{83}Br	8.64E+03	3.63E-04	9.67E-01	1.01E+00	2.87E+04
^{84}Br	1.91E+03	1.04E-04	5.36E-01	5.75E-01	1.60E+04

Table 14-8 – Calculation 4 – U-235 Accidental release for a Public Exposure Time of 24 h at 6.33×10^9 f/s

Nuclide	Eq. 5-2, Eq. 7-1 Time Integrated Exposure			
	Confinement Ventilation		Normal Ventilation	
	Reactor (µCi-h/ml)	Public (µCi-h/ml)	Reactor (µCi-h/ml)	Public (µCi-h/ml)
^{83m}Kr	2.19E-07	2.02E-08	4.20E-07	3.17E-09
^{85m}Kr	5.28E-07	6.67E-08	1.01E-06	7.65E-09
^{85}Kr	1.16E-07	1.99E-08	2.23E-07	1.68E-09
^{87}Kr	1.03E-06	7.80E-08	1.96E-06	1.48E-08
^{88}Kr	1.42E-06	1.56E-07	2.73E-06	2.06E-08
^{89}Kr	1.44E-06	9.54E-09	2.29E-06	1.73E-08
^{131m}Xe	1.59E-08	2.73E-09	3.07E-08	2.32E-10
^{133m}Xe	7.73E-08	1.29E-08	1.49E-07	1.12E-09
^{133}Xe	2.72E-06	4.61E-07	5.24E-06	3.95E-08
^{135m}Xe	4.41E-07	1.07E-08	8.14E-07	6.13E-09
^{135}Xe	2.65E-06	3.87E-07	5.10E-06	3.85E-08
^{137}Xe	2.07E-06	1.60E-08	3.40E-06	2.57E-08
^{138}Xe	2.41E-06	5.47E-08	4.43E-06	3.34E-08
^{131}I	1.20E-06	2.05E-08	2.32E-06	1.75E-08
^{132}I	1.77E-06	1.79E-08	3.40E-06	2.56E-08
^{133}I	2.71E-06	4.33E-08	5.23E-06	3.94E-08
^{134}I	3.11E-06	1.90E-08	5.93E-06	4.47E-08
^{135}I	2.54E-06	3.51E-08	4.89E-06	3.68E-08
^{83}Br	2.21E-07	2.27E-09	4.24E-07	3.19E-09
^{84}Br	1.58E-07	6.82E-10	2.99E-07	2.25E-09

Table 14-8 – Continued

Nuclide	DCF			Eq. 8-4					
				Confinement Ventilation			Normal Ventilation		
	Effective Inhal. rem per µCi-h/ml)	Thyroid Inhal. rem per µCi-h/ml)	Submersr em per µCi-h/ml)	Reactor TEDE (rem)	Reactor Thyroid Dose (rem)	Public TEDE (rem)	Reactor TEDE (rem)	Reactor Thyroid Dose (rem)	Public TEDE (rem)
^{83m}Kr			1.52E-02	3.33E-10	3.33E-10	3.07E-10	6.38E-10	6.38E-10	4.81E-11
^{85m}Kr			1.10E+02	5.82E-06	5.82E-06	7.35E-06	1.12E-05	1.12E-05	8.43E-07
^{85}Kr			1.74E+00	2.02E-08	2.02E-08	3.47E-08	3.88E-08	3.88E-08	2.93E-09
^{87}Kr			5.25E+02	5.39E-05	5.39E-05	4.10E-05	1.03E-04	1.03E-04	7.77E-06
^{88}Kr			1.33E+03	1.89E-04	1.89E-04	2.08E-04	3.64E-04	3.64E-04	2.74E-05
^{89}Kr			1.20E+03	1.73E-04	1.73E-04	1.14E-05	2.75E-04	2.75E-04	2.08E-05
^{131m}Xe			5.48E+00	8.73E-09	8.73E-09	1.49E-08	1.68E-08	1.68E-08	1.27E-09
^{133m}Xe			1.99E+01	1.54E-07	1.54E-07	2.57E-07	2.97E-07	2.97E-07	2.24E-08
^{133}Xe			2.25E+01	6.10E-06	6.10E-06	1.04E-05	1.18E-05	1.18E-05	8.86E-07
^{135m}Xe			2.79E+02	1.23E-05	1.23E-05	2.98E-06	2.27E-05	2.27E-05	1.71E-06
^{135}Xe			1.73E+02	4.59E-05	4.59E-05	6.70E-05	8.84E-05	8.84E-05	6.66E-06
^{137}Xe			1.10E+02	2.28E-05	2.28E-05	1.76E-06	3.74E-05	3.74E-05	2.82E-06
^{138}Xe			7.10E+02	1.71E-04	1.71E-04	3.88E-05	3.14E-04	3.14E-04	2.37E-05
^{131}I	3.95E+04	1.30E+06	2.42E+02	4.74E-02	1.56E+00	8.14E-04	9.15E-02	3.00E+00	6.93E-04
^{132}I	4.57E+02	7.73E+03	1.49E+03	1.07E-03	1.39E-02	3.49E-05	2.06E-03	2.67E-02	4.99E-05
^{133}I	7.02E+03	2.16E+05	3.92E+02	1.91E-02	5.86E-01	3.21E-04	3.69E-02	1.13E+00	2.92E-04

^{134}I	1.58E+02	1.28E+03	1.73E+03	1.03E-03	4.52E-03	3.58E-05	1.96E-03	8.60E-03	8.44E-05
^{135}I	1.47E+03	3.76E+04	1.06E+03	4.01E-03	9.57E-02	8.89E-05	7.73E-03	1.84E-01	9.35E-05
^{83}Br	1.03E+02		5.09E+00	2.30E-05	1.12E-07	2.47E-07	4.41E-05	2.16E-07	3.47E-07
^{84}Br	1.01E+02		1.25E+03	8.76E-05	4.85E-05	2.26E-06	1.65E-04	9.16E-05	7.46E-06
			TOTAL	**7.35E-02**	**2.26E+00**	**1.69E-03**	**1.42E-01**	**4.35E+00**	**1.31E-03**

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

Parameter	Reactor Building		Public Areas
	TEDE (rem)	Thyroid TODE (rem)	TEDE (rem)
Total dose in rem	2.15E-01	6.61E+00	3.00E-03
rem per f/s	1.98E-12	6.08E-11	2.76E-14
Dose limit in rem	1.00E+00	1.00E+01	3.00E-03
Fission rate limit (f/s): Eq. 9-1	5.06E+11	1.64E+11	1.09E+11

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

Table 14-10 lists [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 greater than 96 h.

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)	96 h (s/m^3)	>96 h (s/m^3)
All	30 to 100	up to 12	8.54E-03	3.99E-04	2.31E-04	2.31E-04	1.39E-07	1.56E-06
All	100 to 150	up to 12	2.46E-04	4.73E-05	2.39E-04	2.39E-04	2.73E-06	3.72E-06
All	150 to 5000	up to 30	2.00E-03	4.89E-05	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-03	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	1.54E-03	1.21E-05	2.05E-04	2.05E-04	8.97E-06	5.10E-06
Carroll, Syme	325	12	9.93E-04	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**	**3.99E-04**	**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 - Radiation TEDE for other public areas for accidental releases from encapsulation failure and for accidental releases from vented fueled experiments at the TS radiation dose limit for U-235

Building or Location	Eq. 8-4 Fumign. TEDE 24 h (rem)	Eq. 8-4 Calm Wind TEDE 24 h (rem)	Eq. 8-4 GPM TEDE 24 h (rem)	Eq. 8-4 Fumign. TEDE 2 h (rem)	Eq. 8-4 Fumign. TEDE 24 h (rem)	Eq. 8-4 Calm Wind TEDE 24 h (rem)	Eq. 8-4 GPM TEDE 24 h (rem)	Eq. 8-4 Fumign. TEDE 2 h (rem)
All	3.00E-03	1.63E-04	8.11E-05	2.66E-03	2.78E-04	1.30E-05	7.52E-06	2.46E-04
All	8.64E-05	1.68E-04	8.40E-05	7.66E-05	8.01E-06	1.54E-06	7.78E-06	7.10E-06
All	1.01E-03	1.51E-03	7.55E-04	8.97E-04	9.38E-05	1.59E-06	7.00E-05	8.31E-05
Withers, Mann	1.89E-03	6.08E-05	3.42E-05	1.68E-03	1.75E-04	5.63E-06	3.17E-06	1.55E-04
Broughton, Riddick	1.39E-03	3.29E-05	7.94E-05	1.24E-03	1.29E-04	3.05E-06	7.36E-06	1.15E-04
Patterson, Ricks	1.11E-03	2.04E-05	8.47E-05	9.87E-04	1.03E-04	1.89E-06	7.85E-06	9.15E-05
DH Hill	7.03E-04	7.62E-06	7.55E-04	6.23E-04	6.51E-05	7.06E-07	7.00E-05	5.77E-05
Cox	6.08E-05	5.55E-06	7.62E-05	5.39E-05	5.63E-06	5.14E-07	7.06E-06	4.99E-06
Dabney	5.41E-04	4.29E-06	1.80E-04	4.80E-04	5.01E-05	3.97E-07	1.67E-05	4.44E-05
Hillsborough St.	5.41E-04	4.29E-06	9.10E-05	4.80E-04	5.01E-05	3.97E-07	8.43E-06	4.44E-05
Talley, Reynolds	3.49E-04	4.25E-06	7.20E-05	3.09E-04	3.23E-05	3.94E-07	6.67E-06	2.87E-05
Carroll, Syme	3.76E-04	1.62E-06	5.76E-05	3.33E-04	3.48E-05	1.50E-07	5.34E-06	3.09E-05
North	2.89E-04	1.40E-06	6.18E-05	2.56E-04	2.68E-05	1.30E-07	5.73E-06	2.37E-05
MAXIMUM	**3.00E-03 Encap.**	**1.51E-03 Failure**	**7.55E-04 Accident**	**2.66E-03 Release**	**2.78E-04 Vented**	**1.30E-05 Exp**	**7.00E-05 Accident**	**2.46E-04 Release**

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.
- To determine the TEDE at a specific location, the maximum TEDE may be multiplied by the ratio of the [X/Q] used for a specific location under the listed weather conditions at a given exposure time to the maximum [X/Q] used for the same weather conditions and exposure time.

Supporting Calculations:

Maximum rem at 24 hours using GPM from an encapsulation failure accidental release:

7.55×10^{-4} rem = 3×10^{-3} rem (2.15×10^{-3} s m $^{-3}$ / 8.54×10^{-3} s m $^{-3}$)

Cox Hall at 24 hours using the Calm Wind GPM for a vented experiment accidental release:

5.14×10^{-7} rem = 1.3×10^{-5} rem (1.58×10^{-5} s m $^{-3}$ / 3.99×10^{-4} s m $^{-3}$)

2.78×10^{-4} rem is the maximum TEDE from a vented fueled experiment accidental release. The TEDE is proportional to the mass ratio of the vented fueled experiment to the encapsulated experiment;

$$2.78\times10^{-4} \text{ rem} = 3\times10^{-3} \text{ rem} (5.21\times10^{-3}\text{g} / 5.61\times10^{-2}\text{g})$$

The vented fueled experiment masses are taken from Calculations 1 and 4.

Table 14-12 – Calculation 5 - Radiation TEDE for other public areas for planned releases from vented fueled experiments at the TS radiation dose limit for U-235

Building or Location	Fumigation TEDE 24 h (rem)	Eq 8-4 GPM TEDE 96 h (rem)	GPM TEDE >96 h (rem)
All	3.00E-03	1.94E-07	5.11E-05
All	8.64E-05	3.82E-06	1.22E-04
All	1.01E-03	1.09E-03	3.00E-03
Withers, Mann	1.89E-03	5.88E-14	2.23E-20
Broughton, Riddick	1.39E-03	4.98E-08	3.38E-05
Patterson, Ricks	1.11E-03	1.51E-06	9.38E-05
DH Hill	7.03E-04	1.09E-03	3.00E-03
Cox	6.08E-05	7.95E-06	1.47E-04
Dabney	5.41E-04	2.59E-04	9.41E-04
Hillsbrgh.St	5.41E-04	2.48E-05	2.49E-04
Talley, Reynolds	3.49E-04	1.26E-05	1.67E-04
Carroll, Syme	3.76E-04	2.11E-05	1.77E-04
North	2.89E-04	8.40E-05	3.28E-04
MAXIMUM rem	**3.00E-03 Planned**	**1.09E-03 Vented Experiment**	**3.00E-03 Release**

Talley, Reynolds at >96 hours for vented experiment:

$$1.67\times10^{-4} \text{ rem} = 3\times10^{-3} \text{ rem} (5.10\times10^{-6} \text{ s m}^{-3} / 9.15\times10^{-5} \text{ s m}^{-3})$$

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

The activity of Sr-90, Cs-137, and Pm-147 were calculated using Eq. 9-3.

Table 14-13 gives the 10 CFR Part 37 Category 2 Limit Fractions for U-235 irradiation for the irradiation conditions used for vented fueled experiments in Calculation 1.

For irradiation conditions used in the vented experiment example in Calculation 1, the TS radiation dose criteria is met for 2240 h of exposure from released fission gases and halogens; i.e. a TEDE of 0.003 rem outside the reactor building using [X/Q] for > 96 h and T of 2240 h. The longer irradiation time meets the sum of the activity fractions requirement for 10 CFR Part 37; i.e. a fraction less than 1.

For encapsulated fueled experiments, the TS radiation dose criteria is met for an exposure time of 24 h from an accidental release of fission gases and halogens; i.e. a TEDE of 0.003 rem outside the reactor building using a [X/Q] for 24 h and T of 24 h. Irradiation times greater than 24 h meet the radiation dose of a TEDE of 0.003 rem outside the reactor building from an accidental release. Various irradiation times at various fluence rates were analyzed for compliance with 10 CFR Part 37, including those used in Calculation 4. Results are summarized in Table 14-14.

It is noted that the low radiation dose criteria limits the fission rate, which in turn limits the sample mass and fluence rate. At the fission rate limit for the fluence rates achievable in the experimental facilities,

the activity produced for irradiation times of 24 hours or longer are orders of magnitude below the 10 CFR Part 37 limits.

Table 14-13 – Calculation 6 - 0 CFR Part 37 Category 2 Activity Limit Fractions for Vented Fueled Experiments with U-235 at 1.01×10^{10} f/s for Irradiation Time of 2240 h

		Decay	Fission Yield %			10 CFR Part 37	
Nuclide	Half-life sec	Constant 1/s	Thermal	Non-Thermal	Activity Ci	Category 2 Limit Ci	Fraction of Limit
^{90}Sr	9.07E+08	7.65E-10	5.87	5.60	9.75E-05	3.70E+02	2.63E-07
^{137}Cs	9.47E+08	7.32E-10	3.25	3.76	5.40E-05	2.70E+01	2.00E-06
^{147}Pm	8.26E+07	8.39E-09	2.25	2.14	3.97E-04	1.08E+04	3.67E-08
				TOTAL	**5.48E-04**		**2.30E-06**

Supporting Calculation:

Irradiation time for vented experiments is based on the public radiation dose limit of 0.003 rem.

Calculation 1 was performed for an exposure period of 24 h.

The mass of U-235 at the specified fluence rates in Calculation 1 was 5.21×10^{-3} grams giving a fission rate limit of 1.01×10^{10} f/s.

Radiation dose is related to the product [X/Q] T. At T = 24 h, this product is 0.205 and is matched at

T = 2240 h:

$$0.205 = 24\text{ h } [8.54\times10^{-3}\text{ s m}^{-3}] = 2240\text{ h } [9.15\times10^{-5}\text{ s m}^{-3}]$$

For T = 96 h, the [X/Q] T product is 0.075 and the radiation dose is 0.0011 rem, (0.003 rem x 0.075 / 0.205) and therefore is not a restriction since the radiation dose is less than 0.003 rem.

At T = 2240 h, the radiation dose of 0.003 rem is reached. 2240 h is the limit on T for the vented fueled experiment described in Calculation 1. Irradiation times longer than 2240 h would give a radiation dose greater than 0.003 rem.

At T = 2240 h, the Pm-147 activity is:

$$3.97\times10^{-4}\text{ Ci} = [(5.21\times10^{-3}\text{g})(6.022\times10^{23}\text{ atoms per mole / 235 g per mole})] \times 1\text{ decay per atom}$$

$$\text{x } [(585\times10^{-24})(1\times10^{12})(2.25/100)+(571\times10^{-24})(3\times10^{11})(2.14/100)]\text{ s}^{-1}$$

$$\text{x } [1-e^{-(8.39\text{E-}9\text{ per s})(2240\text{ h})(3600\text{ s/h})}]\ [1\text{Ci} / 3.7\times10^{10}\text{ dps}]$$

At higher fluence rates, the mass of U-235 is reduced to give the same TEDE of 0.003 rem outside the reactor building. As a result, the 10 CFR Part 37 activity fraction is the same.

Table 14-14 – Calculation 6 - 10CFR Part 37 Category 2 Activity Limit Fractions for Encapsulated Fueled Experiments with U-235 at f/s Limits for Various Irradiation Times and Various Fluence Rates

Mass Limit g	f/s Limit	Neutron Fluence $cm^{-2}s^{-1}$		Accident Public TEDE rem	Irradiation Time	Exposure Time Hours	10 CFR Part 37	
		Thermal	Non-thermal				Ci	Fraction
5.61E-02	1.09E+11	1.00E+12	3.00E+11	3.00E-03	24 h	24	6.49E-05	2.67E-07
5.61E-02	1.09E+11	1.00E+12	3.00E+11	3.00E-03	2240 h	24	5.91E-03	2.48E-05
5.61E-02	1.09E+11	1.00E+12	3.00E+11	3.00E-03	1 y	24	2.15E-02	9.60E-05
5.61E-02	1.09E+11	1.00E+12	3.00E+11	3.00E-03	2 y	24	3.94E-02	1.89E-04
5.61E-02	1.09E+11	1.00E+12	3.00E+11	3.00E-03	4 y	24	6.71E-02	3.69E-04
5.61E-02	1.09E+11	1.00E+12	3.00E+11	3.00E-03	6 y	24	8.78E-02	5.41E-04
5.61E-02	1.09E+11	1.00E+12	3.00E+11	3.00E-03	10 y	24	1.18E-01	8.60E-04
3.03E-03	1.08E+11	1.60E+13	8.00E+12	3.00E-03	1 y	24	2.13E-02	9.67E-05
3.03E-03	1.08E+11	1.60E+13	8.00E+12	3.00E-03	5 y	24	7.76E-02	4.60E-04
3.03E-03	1.08E+11	1.60E+13	8.00E+12	3.00E-03	10 y	24	1.17E-01	8.66E-04
7.27E+00	1.10E+11	1.00E+10	1.00E+08	3.00E-03	1 y	24	2.19E-02	9.45E-05
7.27E+00	1.10E+11	1.00E+10	1.00E+08	3.00E-03	10 y	24	1.19E-01	8.46E-04

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

Table 14-15 and Table 14-16 below give the U-235 and Pu-239 Source Terms at the fission rate limit for an accidental release from an encapsulated experiment, i.e. the greater source term of the encapsulated and vented fueled experiments.

Table 14-17 through Table 14-19 give the external dose rates resulting from U-235 at a fission rate of 1.0×10^{11} f/s.

Table 14-20 gives the combined external doses for Pu-239 at a fission rate of 9.9×10^{10} f/s.

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

U-235 external dose rates and dose were higher and are discussed below:

- External dose from the reactor building, overhead plume, and reactor stack were calculated to give 2.7×10^{-4} rem or less to publicly occupied areas outside the reactor building (from Table 14-19, 24 h external dose at 10 m). 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 1.6×10^{-4} rem based on a minimal distance of 1 m to 10 m occupied for 15 minutes (from Table 14-17 initial rem/h at 1 m).
- 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 9.7×10^{-3} rem/h (from Table 14-17 initial dose rate). 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 2.4×10^{-3} rem or less to roof top occupants.

All external radiation doses are below 20% of annual occupational limits to personnel inside the reactor building and 3% of the annual limit for members of the public from an accidental release of fission gases and halogens from fueled experiments.

Table 14-15 – Calculation 7 - U-235 Source Term at 1.1×10^{11} 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)
^{83m}Kr	6.67E-06	3.33E-06	3.49E-07	3.33E-06	1.31E-05	1.89E-04	3.49E-07	1.37E-06	1.97E-05
^{85m}Kr	1.60E-05	9.46E-06	1.15E-06	9.46E-06	3.71E-05	5.36E-04	1.15E-06	4.51E-06	6.51E-05
^{85}Kr	3.50E-06	2.36E-06	3.43E-07	2.36E-06	9.26E-06	1.34E-04	3.43E-07	1.35E-06	1.94E-05
^{87}Kr	3.13E-05	1.38E-05	1.34E-06	1.38E-05	5.42E-05	7.82E-04	1.34E-06	5.27E-06	7.61E-05
^{88}Kr	4.31E-05	2.38E-05	2.69E-06	2.38E-05	9.34E-05	1.35E-03	2.69E-06	1.05E-05	1.52E-04
^{89}Kr	5.37E-05	1.97E-06	1.64E-07	1.97E-06	7.74E-06	1.12E-04	1.64E-07	6.45E-07	9.30E-06
^{131m}Xe	4.82E-07	3.24E-07	4.70E-08	3.24E-07	1.27E-06	1.83E-05	4.70E-08	1.84E-07	2.66E-06
^{133m}Xe	2.34E-06	1.56E-06	2.22E-07	1.56E-06	6.11E-06	8.82E-05	2.22E-07	8.73E-07	1.26E-05
^{133}Xe	8.21E-05	5.50E-05	7.95E-06	5.50E-05	2.16E-04	3.12E-03	7.95E-06	3.12E-05	4.50E-04
^{135m}Xe	1.39E-05	2.21E-06	1.85E-07	2.21E-06	8.68E-06	1.25E-04	1.85E-07	7.24E-07	1.05E-05
^{135}Xe	8.02E-05	5.06E-05	6.67E-06	5.06E-05	1.99E-04	2.87E-03	6.67E-06	2.62E-05	3.78E-04
^{137}Xe	7.46E-05	3.30E-06	2.75E-07	3.30E-06	1.29E-05	1.87E-04	2.75E-07	1.08E-06	1.56E-05
^{138}Xe	7.63E-05	1.13E-05	9.42E-07	1.13E-05	4.43E-05	6.39E-04	9.42E-07	3.70E-06	5.34E-05
^{131}I	3.63E-05	2.44E-05	3.53E-06	2.44E-06	9.57E-06	1.38E-04	3.53E-07	1.39E-06	2.00E-05
^{132}I	5.37E-05	2.83E-05	3.08E-06	2.83E-06	1.11E-05	4.93E-06	3.08E-07	1.21E-06	1.74E-05
^{133}I	8.20E-05	5.37E-05	7.46E-06	5.37E-06	2.11E-05	3.04E-04	7.46E-07	2.93E-06	4.23E-05
^{134}I	9.53E-05	3.58E-05	3.27E-06	3.58E-06	1.40E-05	2.03E-04	3.27E-07	1.28E-06	1.85E-05
^{135}I	7.69E-05	4.74E-05	6.04E-06	4.74E-06	1.86E-05	2.68E-04	6.04E-07	2.37E-06	3.42E-05
^{83}Br	6.71E-06	3.57E-06	3.92E-07	3.57E-07	1.40E-06	2.02E-05	3.92E-08	1.54E-07	2.22E-06
^{84}Br	1.20E-05	3.34E-06	2.88E-07	3.34E-07	1.31E-06	1.89E-05	2.88E-08	1.13E-07	1.63E-06

Supporting Calculations

$\mathbf{C(0) = [A(\infty)/V]}$ 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):

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

For Kr-87:

$C(0) = A(\infty) / V = 7.51\times10^{4}$ µCi / 2.4×10^{9} ml = 3.13×10^{-5} µCi/ml

where k = 1.52×10^{-4} per s + 1.18×10^{-4} per s = 2.7×10^{-4} per s

2 h <C> = $(3.13\times10^{-5}$ µCi/ml) [(1-exp(-2.7×10^{-4}/s*7200s)) / (2.7×10^{-4}/s*7200s)]

= 1.38×10^{-5} µCi/ml

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

Stack Ci = 3.93 <C> at 24 h for Kr-85 = (3.93)(3.43×10^{-7}) Ci = 1.35×10^{-6} Ci

Overhead Line Ci in calm winds = 56.6 <C> at 2 h for Xe-138 = (56.6)(1.13×10^{-5}) Ci = 6.39×10^{-4} Ci

Table 14-16 – Calculation 7 - Pu-239 Source Term at 9.9×10^{10} 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)
^{83m}Kr	3.36E-06	1.68E-06	1.76E-07	1.68E-06	6.60E-06	9.52E-05	1.76E-07	6.90E-07	9.96E-06
^{85m}Kr	6.38E-06	3.77E-06	4.58E-07	3.77E-06	1.48E-05	2.14E-04	4.58E-07	1.80E-06	2.60E-05
^{85}Kr	1.41E-06	9.53E-07	1.39E-07	9.53E-07	3.74E-06	5.40E-05	1.39E-07	5.44E-07	7.86E-06
^{87}Kr	1.12E-05	4.93E-06	4.80E-07	4.93E-06	1.94E-05	2.79E-04	4.80E-07	1.88E-06	2.72E-05
^{88}Kr	1.43E-05	7.86E-06	8.88E-07	7.86E-06	3.09E-05	4.45E-04	8.88E-07	3.49E-06	5.03E-05
^{89}Kr	1.62E-05	5.95E-07	4.96E-08	5.95E-07	2.34E-06	3.37E-05	4.96E-08	1.95E-07	2.81E-06
^{131m}Xe	4.75E-07	3.19E-07	4.63E-08	3.19E-07	1.25E-06	1.81E-05	4.63E-08	1.82E-07	2.62E-06
^{133m}Xe	2.62E-06	1.75E-06	2.49E-07	1.75E-06	6.86E-06	9.89E-05	2.49E-07	9.79E-07	1.41E-05
^{133}Xe	7.83E-05	5.25E-05	7.58E-06	5.25E-05	2.06E-04	2.97E-03	7.58E-06	2.98E-05	4.29E-04
^{135m}Xe	2.12E-05	3.37E-06	2.81E-07	3.37E-06	1.32E-05	1.91E-04	2.81E-07	1.10E-06	1.59E-05
^{135}Xe	8.48E-05	5.36E-05	7.05E-06	5.36E-05	2.10E-04	3.03E-03	7.05E-06	2.77E-05	4.00E-04
^{137}Xe	6.60E-05	2.92E-06	2.43E-07	2.92E-06	1.14E-05	1.65E-04	2.43E-07	9.54E-07	1.38E-05
^{138}Xe	5.65E-05	8.37E-06	6.98E-07	8.37E-06	3.28E-05	4.74E-04	6.98E-07	2.74E-06	3.95E-05
^{131}I	4.32E-05	2.90E-05	4.20E-06	2.90E-06	1.14E-05	1.64E-04	4.20E-07	1.65E-06	2.38E-05
^{132}I	6.00E-05	3.16E-05	3.44E-06	3.16E-06	1.24E-05	1.79E-04	3.44E-07	1.35E-06	1.95E-05
^{133}I	7.78E-05	5.09E-05	7.07E-06	5.09E-06	2.00E-05	2.88E-04	7.07E-07	2.78E-06	4.01E-05
^{134}I	8.20E-05	3.08E-05	2.81E-06	3.08E-06	1.21E-05	1.74E-04	2.81E-07	1.10E-06	1.59E-05
^{135}I	7.18E-05	4.43E-05	5.65E-06	4.43E-06	1.74E-05	2.51E-04	5.65E-07	2.22E-06	3.20E-05
^{83}Br	3.37E-06	1.79E-06	1.97E-07	1.79E-07	7.03E-07	1.01E-05	1.97E-08	7.71E-08	1.11E-06
^{84}Br	4.89E-06	1.37E-06	1.17E-07	1.37E-07	5.36E-07	7.74E-06	1.17E-08	4.61E-08	6.65E-07

Table 14-17 – Calculation 7 - External dose rates using Microshield and calculated dose from the reactor building for U-235 at 1.1×10^{11} f/s

Distance m	Initial rem/h	2 h TEDE rem/h	24 h TEDE rem/h	2 h TEDE rem	24 h TEDE rem
1	6.34E-04	2.31E-04	2.48E-05	4.62E-04	5.96E-04
10	2.71E-04	1.00E-04	1.08E-05	2.00E-04	2.59E-04
20	1.13E-04	4.19E-05	4.50E-06	8.38E-05	1.08E-04
30	5.82E-05	2.16E-05	2.33E-06	4.33E-05	5.59E-05
40	3.47E-05	1.28E-05	1.38E-06	2.56E-05	3.31E-05
50	2.24E-05	8.28E-06	8.92E-07	1.66E-05	2.14E-05
Roof	9.71E-03				

Notes:

- 24 h dose at 30 m = 5.59×10^{-5} rem = (24 h) (2.48×10^{-5} 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 and dose calculated using Microshield from overhead plume and reactor stack for U-235 at 1.1×10^{11} f/s

Location x,y,z (m)	2 h Average Stack Line TEDE Rate (rem/h)	2 h Average Overhead Line TEDE Rate (rem/h)	2 h Average Stack and Plume TEDE (rem)	24 h Average Stack Line TEDE Rate (rem/h)	24 h Average Overhead Line TEDE Rate (rem/h)	24 h Average Stack and Plume TEDE (rem)
30,0,0	1.81E-07	2.18E-06	4.73E-06	1.96E-08	2.37E-07	6.15E-06
40,0,0	1.15E-07	2.28E-06	4.79E-06	1.24E-08	2.47E-07	6.23E-06
50,0,0	7.77E-08	2.31E-06	4.77E-06	8.41E-09	2.50E-07	6.21E-06
10,0,12	1.52E-06	3.54E-06	1.01E-05	1.65E-07	3.85E-07	1.32E-05
20,0,12	4.97E-07	4.16E-06	9.31E-06	5.38E-08	4.50E-07	1.21E-05
30,0,12	2.38E-07	4.46E-06	9.40E-06	2.59E-08	4.84E-07	1.22E-05
40,0,12	1.37E-07	4.60E-06	9.47E-06	1.50E-08	5.01E-07	1.24E-05
50,0,12	8.79E-08	4.67E-06	9.51E-06	9.50E-09	5.04E-07	1.23E-05

Table 14-19 – Calculation 7 - Combined external doses (sum of reactor building, overhead plume, and stack) for U-235 at 1.1×10^{11} f/s

Location x,y,z (m)	External Dose in 2 h TEDE (rem)	External Dose in 24 h TEDE (rem)
10,0,12	2.1E-04	2.7E-04
20,0,12	9.3E-05	1.2E-04
30,0,12	5.3E-05	6.8E-05
40,0,12	3.5E-05	4.5E-05
50,0,12	2.6E-05	3.4E-05

Note: 2 h dose at 50 m = 2.6×10^{-5} rem = $(1.66\times10^{-5} + 9.51\times10^{-6})$ rem

Table 14-20 – Calculation 7 - Combined external doses (sum of reactor building, overhead plume, and stack) for Pu-239 at 9.9×10^{10} f/s

Location x,y,z (m)	External Dose in 2 h TEDE (rem)	External Dose in 24 h TEDE (rem)
10,0,12	1.6E-04	2.1E-04
20,0,12	7.1E-05	9.3E-05
30,0,12	4.0E-05	5.2E-05
40,0,12	2.6E-05	3.4E-05
50,0,12	1.9E-05	2.5E-05

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)

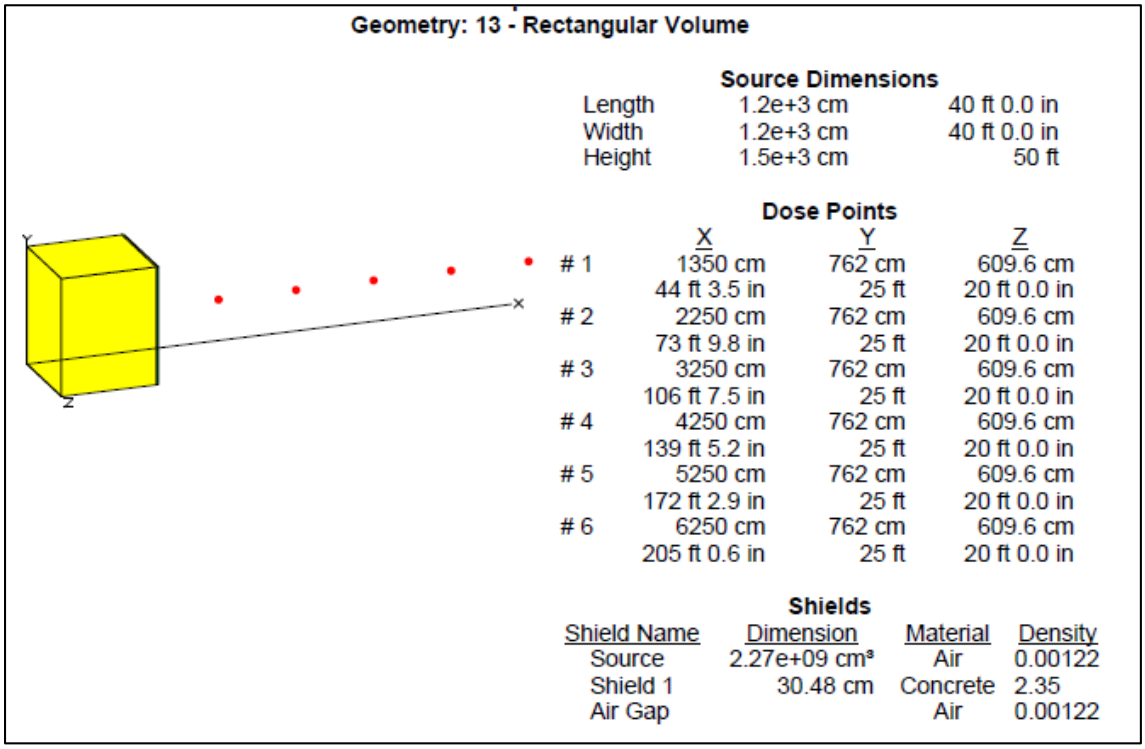
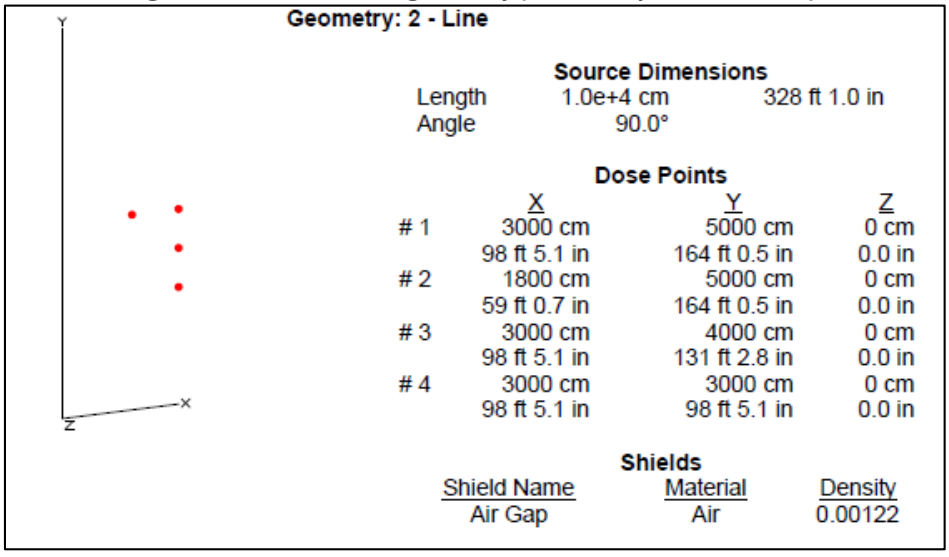


Figure 14-2 – Line source geometry (overhead plume is shown)



CALCULATION 8: External dose rates from beamtube exhaust duct

External dose rates in the beamtube exhaust duct were calculated for a vented fueled experiment with an exhaust rate of 3 lpm (50 ml/s) and a decay time of 30 min (decay volume of 90 liters). For the conditions given in Calculations 1 and 2, the maximum exposure time is 2240 hours. Halogen filter efficiency is 99%.

Table 14-21 and Table 14-22 give the beamtube exhaust duct activity and total dose resulting from U-235 with a fission rate of 1.01×10^{10} f/s with 2240 h exposure. The total activity in the beamtube exhaust for a vented experiment is 135 µCi for U-235. External dose rates for occupied areas (3 m or greater) from the beamtube exhaust duct are 1.8×10^{-6} rem/h or less for U-235. The total dose for 2240 h exposure is 4×10^{-3} rem or less for U-235.

Table 14-23 and Table 14-24 give the beamtube exhaust duct activity and total dose resulting from Pu-239 with a fission rate of 1.45×10^{10} f/s with 2240 h exposure. The total activity in the beamtube exhaust for a vented experiment is 168 µCi for Pu-239. External dose rates for occupied areas (3 m or greater) from the beamtube exhaust duct are 1.6×10^{-6} rem/h or less for Pu-239. The total dose for 2240 h exposure is 4×10^{-3} rem or less for Pu-239.

Table 14-21 – Calculation 8 - Beamtube Exhaust Duct Activity resulting from U-235 at 1.01×10^{10} f/s

Nuclide	Eq. 5-8 Decayed, Filtered Vented Release Rate q, (µCi/s)	Eq. 8-3 Beamtube Duct A(d), (µCi)
^{83m}Kr	6.86E-01	2.74E+00
^{85m}Kr	1.83E+00	7.33E+00
^{85}Kr	4.34E-01	1.73E+00
^{87}Kr	2.95E+00	1.18E+01
^{88}Kr	4.73E+00	1.89E+01
^{89}Kr	9.04E-03	3.62E-02
^{131m}Xe	5.96E-02	2.38E-01
^{133m}Xe	2.87E-01	1.15E+00
^{133}Xe	1.01E+01	4.05E+01
^{135m}Xe	4.43E-01	1.77E+00
^{135}Xe	9.56E+00	3.82E+01
^{137}Xe	4.00E-02	1.60E-01
^{138}Xe	2.16E+00	8.65E+00
^{131}I	4.49E-02	1.79E-01
^{132}I	5.72E-02	2.29E-01
^{133}I	9.99E-02	4.00E-01
^{134}I	7.95E-02	3.18E-01
^{135}I	9.03E-02	3.61E-01
^{83}Br	7.19E-03	2.88E-02
^{84}Br	7.70E-03	3.08E-02
TOTAL	**3.37E+01**	**1.35E+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 - Beamtube Exhaust Duct Dose Rates from Microshield and Calculated Dose resulting from U-235 at 1.01×10^{10} f/s and 2240 hour exposure

Distance m	TEDE Rate (rem/h)	TEDE (rem)
1	6.56E-06	1.5E-02
2	2.99E-06	6.7E-03
3	1.81E-06	4.1E-03
4	1.23E-06	2.8E-03
5	8.97E-07	2.0E-03
6	6.81E-07	1.5E-03
7	5.34E-07	1.2E-03
8	4.30E-07	9.6E-04
9	3.52E-07	7.9E-04
10	2.94E-07	6.6E-04

Table 14-23 – Calculation 8 - Beamtube Exhaust Duct Activity resulting from Pu-239 at 1.45×10^{10} f/s

Nuclide	Eq 5-8 Decayed, Filtered Vented Release Rate q (µCi/s)	Eq 8-3 Beamtube Duct A(d) (µCi)
^{83m}Kr	5.44E-01	2.17E+00
^{85m}Kr	1.15E+00	4.59E+00
^{85}Kr	2.75E-01	1.10E+00
^{87}Kr	1.66E+00	6.63E+00
^{88}Kr	2.46E+00	9.83E+00
^{89}Kr	4.29E-03	1.72E-02
^{131m}Xe	9.24E-02	3.69E-01
^{133m}Xe	5.07E-01	2.03E+00
^{133}Xe	1.52E+01	6.08E+01
^{135m}Xe	1.06E+00	4.24E+00
^{135}Xe	1.59E+01	6.36E+01
^{137}Xe	5.55E-02	2.22E-01
^{138}Xe	2.52E+00	1.01E+01
^{131}I	8.39E-02	3.35E-01
^{132}I	1.00E-01	4.02E-01
^{133}I	1.49E-01	5.95E-01
^{134}I	1.07E-01	4.30E-01
^{135}I	1.33E-01	5.31E-01
^{83}Br	5.67E-03	2.27E-02
^{84}Br	4.95E-03	1.98E-02
TOTAL	**4.20E+01**	**1.68E+02**

Table 14-24 – Calculation 8 - Beamtube Exhaust Duct Dose Rates from Microshield and Calculated Dose resulting from Pu-239 at 1.45×10^{10} f/s and > 96 hour exposure

Distance (m)	TEDE Rate (rem/h)	TEDE (rem)
1	5.82E-06	1.3E-02
2	2.66E-06	6.0E-03
3	1.61E-06	3.6E-03
4	1.10E-06	2.5E-03
5	8.00E-07	1.8E-03
6	6.08E-07	1.4E-03
7	4.77E-07	1.1E-03
8	3.84E-07	8.6E-04
9	3.15E-07	7.1E-04
10	2.63E-07	5.9E-04

All external radiation doses are below 20% of annual occupational limits to personnel inside the reactor building and 3% of the annual limit for members of the public from planned releases of fission gases and halogens from vented fueled experiments.

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 2.0×10^{6} f/s is equivalent to 1.0×10^{-6} g of U-235. Results indicate 2.0×10^{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 2.0×10^{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.

At 2.0×10^{6} f/s, Sr-90, Cs-137, and Pm-147 saturation activity is less than 7 µCi.

Experiments with uranium equal to or greater than 2.0×10^{6} f/s are therefore defined as a fueled experiment.

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

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	U-235		Target atoms, N	2.69E+15	atoms
Mass	1.05E-06	g	Thermal fission rate	1.57E+06	f/s
Mass Number, A	235	g/mol	Non-thermal fission rate	4.61E+05	f/s
Sigma thermal	585	barns	Total fission rate	2.03E+06	f/s
Sigma non-thermal	571	barns	Total fissions		
X/Q	8.54E-03	s/m^3	Reactor volume	2.40E+09	ml
Thermal flux	1.00E+12	cm^2/s	F confinement	0.283	m^3/s
Non-thermal flux	3.00E+11	cm^2/s	v confinement	1.18E-04	1/s
Max Irradiation time	1.83E+07	s	F normal	0.883	m^3/s
			v normal	3.92E-04	1/s
					s
					s
			NG reactor correction	0.1	s
Public exposure time	24	hours			

Table 14-25 – Continued

ISOTOPIC DATA					
Nuclide	**Half-Life (sec)**	**Decay Constant (1/s)**	**Cumulative Yield %**		**Eq. 2-1 Saturation Activity (μCi)**
			Thermal Fission	**Non-Thermal Fission**	
^{83m}Kr	6.70E+03	1.04E-04	*5.36E-01*	*5.75E-01*	3.00E-01
^{85m}Kr	1.61E+04	4.30E-05	*1.29E+00*	*1.36E+00*	7.18E-01
^{85}Kr	3.39E+08	2.05E-09	*2.83E-01*	*2.96E-01*	1.57E-01
^{87}Kr	4.57E+03	1.52E-04	*2.56E+00*	*2.54E+00*	1.41E+00
^{88}Kr	1.02E+04	6.78E-05	*3.55E+00*	*3.43E+00*	1.94E+00
^{89}Kr	1.89E+02	3.67E-03	*4.51E+00*	*3.97E+00*	2.41E+00
^{131m}Xe	1.03E+06	6.74E-07	*4.05E-02*	*3.54E-02*	2.16E-02
^{133m}Xe	1.89E+05	3.66E-06	*1.89E-01*	*1.97E-01*	1.05E-01
^{133}Xe	4.53E+05	1.53E-06	*6.70E+00*	*6.71E+00*	3.69E+00
^{135m}Xe	9.18E+02	7.55E-04	*1.10E+00*	*1.26E+00*	6.25E-01
^{135}Xe	3.28E+04	2.12E-05	*6.54E+00*	*6.58E+00*	3.60E+00
^{137}Xe	2.29E+02	3.02E-03	*6.13E+00*	*5.98E+00*	3.35E+00
^{138}Xe	8.46E+02	8.19E-04	*6.30E+00*	*6.00E+00*	3.43E+00
^{131}I	6.93E+05	1.00E-06	*2.89E+00*	*3.22E+00*	1.63E+00
^{132}I	8.26E+03	8.39E-05	*4.31E+00*	*4.66E+00*	2.41E+00
^{133}I	7.49E+04	9.26E-06	*6.70E+00*	*6.70E+00*	3.69E+00
^{134}I	3.16E+03	2.20E-04	*7.83E+00*	*7.63E+00*	4.28E+00
^{135}I	2.37E+04	2.93E-05	*6.28E+00*	*6.27E+00*	3.45E+00
^{83}Br	8.64E+03	8.02E-05	*5.40E-01*	*5.76E-01*	3.01E-01
^{84}Br	1.91E+03	3.63E-04	*9.67E-01*	*1.01E+00*	5.37E-01

Table 14-26 – Calculation 9 - Time Integrated Exposures and Doses from Irradiation of uranium at 2.0×10^6 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	rem per h/ml Effective Inhalation DCF	rem per µCi-h/ml Thyroid Inhalation DCF	rem per µCi-h/ml Submersion DCF	Eq 8-4 Reactor Normal TEDE rem	Eq 8-4 Reactor Normal Thyroid TODE rem	Eq 8-4 Public Normal TEDE rem
^{83m}Kr	3.00E-09	2.26E-11			1.52E-02	4.55E-12	4.55E-12	3.43E-13
^{85m}Kr	7.18E-09	5.41E-11			1.10E+02	7.92E-08	7.92E-08	5.97E-09
^{85}Kr	1.57E-09	1.19E-11			1.74E+00	2.73E-10	2.73E-10	2.06E-11
^{87}Kr	1.41E-08	1.06E-10			5.25E+02	7.38E-07	7.38E-07	5.57E-08
^{88}Kr	1.94E-08	1.46E-10			1.33E+03	2.58E-06	2.58E-06	1.95E-07
^{89}Kr	2.41E-08	1.82E-10			1.20E+03	2.90E-06	2.90E-06	2.18E-07
^{131m}Xe	2.16E-10	1.63E-12			5.48E+00	1.18E-10	1.18E-10	8.93E-12
^{133m}Xe	1.05E-09	7.91E-12			1.99E+01	2.09E-09	2.09E-09	1.58E-10
^{133}Xe	3.69E-08	2.78E-10			2.25E+01	8.28E-08	8.28E-08	6.24E-09
^{135m}Xe	6.25E-09	4.71E-11			2.79E+02	1.74E-07	1.74E-07	1.31E-08
^{135}Xe	3.60E-08	2.71E-10			1.73E+02	6.24E-07	6.24E-07	4.70E-08
^{137}Xe	3.35E-08	2.53E-10			1.10E+02	3.69E-07	3.69E-07	2.78E-08
^{138}Xe	3.43E-08	2.58E-10			7.10E+02	2.43E-06	2.43E-06	1.83E-07
^{131}I	1.63E-08	1.23E-10	3.95E+04	1.30E+06	2.42E+02	6.44E-04	2.11E-02	4.88E-06
^{132}I	2.41E-08	1.82E-10	4.57E+02	7.73E+03	1.49E+03	1.46E-05	1.90E-04	3.55E-07
^{133}I	3.69E-08	2.78E-10	7.02E+03	2.16E+05	3.92E+02	2.60E-04	7.95E-03	2.06E-06
^{134}I	4.28E-08	3.23E-10	1.58E+02	1.28E+03	1.73E+03	1.42E-05	6.22E-05	6.10E-07
^{135}I	3.45E-08	2.60E-10	1.47E+03	3.76E+04	1.06E+03	5.46E-05	1.30E-03	6.60E-07
^{83}Br	3.01E-09	2.27E-11	1.03E+02		5.09E+00	3.13E-07	1.53E-09	2.46E-09
^{84}Br	5.37E-09	4.05E-11	1.01E+02		1.25E+03	1.21E-06	6.73E-07	5.48E-08

*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 2.0×10^6 f/s

	Reactor Building	Public Areas
Total TEDE (rem)	0.001	9.37E-06
TEDE Limit (rem)	0.001	0.001
rem dose per f/s	4.91E-10	4.61E-12
Eq 9-1 Fission rate limit (f/s)	2.04E+06	2.17E+08

Note: Sr-90, Cs-137, and Pm-147 saturation activity is less than 7 µCi.

CALCULATION 10: Radiation and Experiment Monitor Channel Setpoint Calculations

Setpoint bases:

Setpoints for the reactor stack radiation monitors are based on the maximum concentrations of Ar-41 and fission gases or halogens expected during normal operations. Vented fueled experiment releases within TS limits are normal and would not trip any monitor setpoints. Setpoints are below releases that exceed a TEDE of 0.003 rem to members of the public outside the reactor building from fueled experiment accidents. The TEDE of 0.003 rem occurring over longer release periods is monitored using facility procedures on a frequent basis, typically monthly. Setpoints for the vented fueled experiment exhaust gas radiation monitor are based on fission gas releases at the TS limit. The alarm setpoint for the vented fueled experiment gas flow rate monitor is set just above the required flow rate for each experiment to assure that the minimum decay time criteria is met. A controlled, planned release from vented experiments should not cause alarms or annunciation from radiation monitors.

- The vented fueled experiment exhaust gas radiation monitor is used to monitor vented fueled experiments.
- The stack gas and stack exhaust radiation monitors are used to monitor fission gases from vented fueled experiments and Ar-41 releases.
- The stack particulate radiation monitor with a particulate filter and radioiodine sample cartridge is used to monitor halogen releases from vented fueled experiments.

Radiation monitor response is based on the release concentration at the experiment exhaust and reactor stack. Concentration is used for radioactive gas detection while accumulated activity on the radioiodine cartridge is used for detection of halogens.

Release concentrations at the reactor stack are lower for vented fueled experiments than for accidental releases. Therefore, vented fueled experiments and normal Ar-41 releases are used to determine radiation monitor setpoints for the reactor stack while a vented fueled experiment is performed.

Released concentration for vented fueled experiments:

Vented fueled experiment public area and reactor stack concentrations may be calculated from the release rate (Eq 5-7 and Eq 5-8) or time-integrated exposure (Eq 7-4) and the exposure time of 24 h. 24 h exposure in fumigation weather conditions gave the limiting fission rates from vented fueled experiments that resulted in an off-site radiation dose of 0.003 rem.

$$Public\ C = \Psi_p / T \qquad \text{EQ. 14-1}$$

$$Stack\ C = \Psi_p / (F[X/Q]T) \qquad \text{EQ. 14-2}$$

where, C is the concentration in µCi/ml for an individual radionuclide

Ψ_p is the time integrated exposure in µCi-h/ml for members of the public in µCi-h/ml from EQ 7-4

[X/Q] is atmospheric dispersion parameter of 8.5410^{-3} s/m^3

T is 24 h

F is reactor stack flow rate of 0.883 m^3/s for normal ventilation

Vented fueled experiment EC fraction:

Using 10 CFR Part 20 Appendix B EC values, the vented fueled experiment EC fractions in the reactor stack were calculated for each radionuclide:

$$Stack\ EC\ Fraction = Stack\ C/EC \qquad \text{EQ. 14-3}$$

EC-h Calculation:

Public EC-h were calculated for each radionuclide released using Eq 14-4:

$$Public\ EC \cdot h = Stack\ EC\ Fraction \times (F[X/Q]T) \qquad \text{EQ. 14-4}$$

Public EC-h is related to public radiation dose:

$$Public\ Dose\ in\ rem = 0.1\ rem \times (Public\ EC \cdot h/8760\ h) \qquad \text{EQ. 14-5}$$

e.g. 2.27×10^{-3} rem = 0.1x [199/8760] rem

The values for Public C, Stack C, Stack EC fraction, and Public EC-h were summed for (1) all fission gases and (2) all halogens. The EC for (1) fission gases and (2) halogens from vented fueled experiments, EC_{vfe} were calculated using the summed values for Stack C [Eq 14-2] and Stack EC Fraction [Eq 14-3]:

$$EC_{vfe} = {\sum Stack\ C}/{\sum Stack\ EC\ Fraction} \qquad \text{EQ. 14-6}$$

Results for the example U-235 vented fueled experiment from Calculation 1 are given in Table 14-28.

Stack Gas Radiation Monitor Setpoints

The alarm setpoint for the stack gas and stack exhaust radiation monitors are based on detecting abnormal releases of Ar-41 and fission gases from vented fueled experiments and not exceeding the Emergency Action Level (EAL) of 50 EC fractions for 24 hours, or 1200 EC-h, with an associated radiation dose of approximately 0.015 rem in 24 hours. The alert setpoints are set below the alarm setpoints and detect abnormal releases of Ar-41 and fission gases from vented fueled experiments. Ar-41 concentrations based on the EAL for the alarm setpoint is 9.3×10^{-5} µCi/ml. Maximum Ar-41 concentration from normal operations is 2.7×10^{-5} µCi/ml. The concentration calculated in Eq 14-2 for the stack gas and stack exhaust radiation monitors is used as the setpoint for vented fueled experiments. For the example of U-235 given in Table 14-28, the fission gas concentration for the alarm setpoint is:

$$3.77\text{x}10^{-5}\ \mu\text{Ci}/ml = (6.83\text{x}10^{-6}\ \mu\text{Ci} - \text{h}/ml)/\ [(0.00854\ \text{s/m}^3)(0.883\ \text{m}^3/\text{s})(24\ \text{h})] \qquad \text{EQ. 14-7}$$

The stack gas and stack exhaust radiation monitors provide count rates that are proportional to the concentration. The detector efficiency is dependent on radiation type, yield, and energy. Therefore, the alarm setpoints are based on the count rate from the vented fueled experiment and normal amounts of Ar-41; or the total count rate for 2.7×10^{-5} µCi/ml of Ar-41 and 3.77×10^{-5} µCi/ml of fission gases. The setpoints are also less than the EAL concentration of 9.3×10^{-5} µCi/ml of Ar-41 and initial concentrations of fission gases from a fueled experiment accident giving a TEDE of 0.003 rem in 24 h.

Table 14-28 – Calculation 10 - Example U-235 Vented Fueled Experiment Release Data

Nuclide	Ψp Time Integrated Exposure µCi-h/ml	Eq 14-1 Public C µCi/ml	Eq 14-2 Stack C µCi/ml	10 CFR 20 App B EC µCi/ml	Eq 14-3 Stack EC Fraction	Eq-14-5 Public EC-h
^{83m}Kr	1.41E-07	5.86E-09	7.76E-07	5.E-05	1.55E-02	2.81E-03
^{85m}Kr	3.75E-07	1.56E-08	2.07E-06	1.E-07	2.07E+01	3.75E+00
^{85}Kr	8.89E-08	3.70E-09	4.91E-07	7.E-07	7.01E-01	1.27E-01
^{87}Kr	6.04E-07	2.52E-08	3.34E-06	2.E-08	1.67E+02	3.02E+01
^{88}Kr	9.69E-07	4.04E-08	5.35E-06	9.E-09	5.95E+02	1.08E+02
^{89}Kr	1.85E-09	7.72E-11	1.02E-08	1.E-08	1.08E+00	1.95E-01
^{131m}Xe	1.22E-08	5.09E-10	6.75E-08	2.E-06	3.37E-02	6.11E-03
^{133m}Xe	5.89E-08	2.46E-09	3.26E-07	6.E-07	5.43E-01	9.82E-02
^{133}Xe	2.08E-06	8.65E-08	1.15E-05	5.E-07	2.30E+01	4.15E+00
^{135m}Xe	9.08E-08	3.78E-09	5.02E-07	4.E-08	1.25E+01	2.27E+00
^{135}Xe	1.96E-06	8.16E-08	1.08E-05	7.E-08	1.55E+02	2.80E+01
^{137}Xe	8.19E-09	3.41E-10	4.53E-08	1.E-07	4.53E-01	8.19E-02
^{138}Xe	4.43E-07	1.85E-08	2.45E-06	2.E-08	1.22E+02	2.22E+01
^{131}I	9.19E-09	3.83E-10	5.08E-08	2.E-10	2.54E+02	4.60E+01
^{132}I	1.17E-08	4.89E-10	6.48E-08	2.E-08	3.24E+00	5.86E-01
^{133}I	2.05E-08	8.53E-10	1.13E-07	1.E-09	1.13E+02	2.05E+01
^{134}I	1.63E-08	6.79E-10	9.00E-08	6.E-08	1.50E+00	2.72E-01
^{135}I	1.85E-08	7.71E-10	1.02E-07	6.E-09	1.70E+01	3.08E+00
^{83}Br	1.47E-09	6.14E-11	8.14E-09	9.E-08	9.04E-02	1.64E-02
^{84}Br	1.58E-09	6.57E-11	8.72E-09	8.E-08	1.09E-01	1.97E-02
Fission Gas total	6.83E-06	2.85E-07	3.77E-05	Eq 14-6 3.44E-08	1098	199
Halogen total	7.93E-08	3.30E-09	4.38E-07	Eq 14-6 1.13E-09	389	70

Stack Particulate Radiation Monitor Setpoints:

The alarm setpoint for the stack particulate radiation monitor is based on an abnormal release of halogens from a vented fueled experiment and not exceeding the EAL of 100 EC fractions for 24 hours, or 2400 EC-h with an associated radiation dose of approximately 0.015 rem in 24 hours. The limiting radionuclide used for the EAL is Co-60, which has an EC of 5×10^{-11} µCi/ml. Co-60 concentrations based on the EAL for the alarm and alert setpoints are 6.63×10^{-7} µCi/ml. The stack concentration calculated in Eq. 14-2 is used for the halogen based setpoint for vented fueled experiments. For the example of U-235 given in Table 14-28, the halogen concentration for the alarm setpoint is:

$$4.38 \times 10^{-7}\ \mu\mathrm{Ci}/ml = (7.98 \times 10^{-8}\ \mu\mathrm{Ci-h}/ml)/\left[(0.00854\ \mathrm{s/m^3})(0.883\ \mathrm{m^3/s})(24\ \mathrm{h})\right] \quad \text{EQ. 14-8}$$

The stack particulate radiation monitor detector provides a count rate that is proportional to the activity present on the sampling filter. The detector efficiency is dependent on radiation type, yield, and energy. Therefore, the alarm setpoint is based on the count rate from the halogens released from a vented fueled experiment and less than the EAL based on for Co-60 particulates.

Table 14-29 summarizes the example U-235 vented fueled experiment stack gas and particulate radiation monitor setpoints.

Table 14-29 – Calculation 10 - Example U-235 Vented Fueled Experiment Set Points

Nuclide	EC µCi/ml	EC-h	Stack EC Fraction Set Point	Stack Concentration Set Point, µCi/ml
Ar-41	1.00E-08	1200	6631	6.63E-5
Fission Gas	3.44E-08	199	1098	3.77E-5
Co-60	5.00E-11	2400	13261	6.63E-7
Halogens	1.13E-09	70	389	4.38E-7

Vented fueled experiment exhaust gas radiation monitor setpoints:

Dilution from the normal reactor ventilation exhaust and the vented experiment exhaust is used to determine the vented fueled experiment exhaust gas radiation monitor setpoint.

EQ 5-5 and Table 14-28 are used to calculate the vented fueled experiment exhaust gas radiation monitor setpoint based on fission gas release:

$$\text{Vented Fueled Experiment Exhaust Gas Setpoint } EC_{vfe} = 1098\ EC_{vfe} \times [(p+F)/p] \qquad \text{EQ. 14-9}$$

For the U-235 example given in Calculation 1, the alarm setpoint would be:

$$1.9\ E7\ EC_{vfe} = (1098\ EC_{vfe} \times [(50\times10^{-6} + 0.883)\ m^3/s\ /\ (50\times10^{-6} m^3/s)] \qquad \text{EQ. 14-10}$$

For the U-235 example, this equates to an experiment exhaust concentration of:

$$\text{Vented experiment exhaust gas C} = \text{Stack C} \times [(50\times10^{-6} + 0.883)\ m^3/s\ /\ (50\times10^{-6} m^3/s)]$$
$$0.67\ \mu Ci/ml = 3.77\times10^{-5}\ \mu Ci/ml \times [(50\times10^{-6} + 0.883)\ m^3/s\ /\ (50\times10^{-6} m^3/s)] \qquad \text{EQ. 14-11}$$

Vented fueled experiment exhaust gas flow rate monitor setpoint:

The alarm setpoint for the vented fueled experiment gas flow rate monitor is set at the maximum flow rate determined for each experiment which assures that the minimum decay time criterion is met. For the U-235 example given in Calculation 1, a decay time of 30 minutes is required, which correlates to a maximum flow rate of 3 lpm. The vented fueled experiment gas flow rate monitor setpoint would therefore be set at 3 lpm for this experiment to assure that the minimum decay time of 30 minutes is met.

Setpoints for Accidental Releases

Setpoints for accidental releases from a vented or encapsulated fueled experiments are based on initial concentrations in the reactor stack without any filtration or decay.

Using the setpoint calculations described above for initial concentrations for an accidental release from a fueled experiment as reported in Table 14-15 gives the following:

Table 14-30 – Accident Setpoints

	Stack C(0) µCi/ml	EC µCi/ml	Stack EC Fraction
Fission Gas total	4.84E-04	2.63E-08	18402
Halogen total	3.63E-04	1.29E-09	280829

The stack concentration values for accidental releases exceed those for the routine release from a vented fueled experiment and the EAL based setpoints for Ar-41 and Co-60 given in Table 14-29.

For this example, a vented fueled experiment accident release has a public dose less than 3×10^{-4} rem, an initial fission gas concentration of 4.5×10^{-5} µCi/ml, and a halogen concentration of 3.4×10^{-5} µCi/ml. Both concentrations are above the values given in Table 14-29.

Calculation 11: Calculations for Additional Fissionable Materials

Calculations for other fissionable materials were made as described in this analysis. The amounts of specific fissionable materials present in a fueled experiment are determined before the experiment begins. Mixtures of fissionable materials are commonly used.

Common fissionable materials potentially present in fueled experiments include the following:

Uranium (U):

U-234, U-235 and U-238 are present in natural abundance, U-235 enriched U, or depleted U. U-236 and U-239 are not produced in significant amounts from the activation of U. U-236 is produced by activation of U-235. Activation of U-238 produces U-239 which beta decays to Np-239 and then beta decays to Pu-239.

Thorium (Th):

U-232 and U-233 are produced from (n,γ) and (n,2n) nuclear reactions with Th followed by beta decay. Activation of Th-232 followed by beta decay leads to U-233. Th-232 is the major nuclide of Th and activates to Th-233 with decay to U-233. U-232 is produced as a side product from nuclear reactions with other nuclides of Th or from (n,2n) reactions with U-233.

Neptunium (Np):

Np-237 is a long-lived radionuclide. Np-237 undergoes activation to produce Np-238 which beta decays to produce Pu-238.

Plutonium (Pu):

Pu-239 is commonly present in samples of Pu. Other isotopes of Pu are typically present in lower amounts. The fission reaction cross-sections for Pu-239 are significantly larger than those for Pu-238, Pu-240, and Pu-242 and lower than those for Pu-241.

Example Calculations:

Results for vented experiments and encapsulated experiments are provided in Table 14-31 for a range of different fissionable materials. Conditions for vented experiments were at an exhaust flow rate of 3 lpm and decay time of 30 minutes, and fluence rates of 1×10^{12} thermal and 3×10^{11} non-thermal were used for both vented and encapsulated experiments. In all experimental cases, the limiting TS dose criteria of a maximum TEDE of 0.003 rem to members of the public outside the reactor building is met. The data in

Table 14-31 was used to determine f/s per g and rem/g and rem per f/s in public areas as given in Table 14-32.

Calculations for other experimental conditions were made as described in this analysis with examples provided in Table 14-33. These include different release rates and decay times for vented experiments, different thermal to non-thermal fluence rate ratios, and different maximum neutron fluxes correlating to 1 and 2 MW power levels. In all experimental cases, the limiting TS dose criteria of a maximum TEDE of 0.003 rem to members of the public outside the reactor building is met. Supporting data used in these calculations is given in Table 14-34 through Table 14-37.

For mixtures, the most restrictive fissionable material present that produces greater than 50 percent of radiation dose is used to determine the fission rate limit for that particular mixture.

Restrictive fissionable materials were determined from radiation dose per unit mass, mass, and fission rate given in Table 14-32 for common mixtures. Results for these mixtures are as follows:

Mixture	Restrictive Material
U-232, U-233	U-233
U-234, U-235, U-236, U-238, Pu-239	U-235
Np-237, Np-238, Pu-238	Np-237
Pu 238, Pu-239, Pu-240, Pu-241, Pu-242	Pu-239

For example, dose to the public from an encapsulation failure accident from a 1 mg sample of natural U is:

U-234: 2.78×10^{-5} rem/g x 0.00054 × 0.001g = 1.50×10^{-11} rem

U-235: 5.35×10^{-2} rem/g x 0.0072 × 0.001g = 3.85×10^{-7} rem

U-238: 6.38×10^{-6} rem/g x 0.992746 × 0.001g = 6.34×10^{-9} rem

Pu-239: 7.5×10^{-2} rem/g x 0.992746 × 1×10^{-7}g = 7.5×10^{-9} rem

For the 1 mg of natural U sample, U-235 gives > 98% of the radiation dose to the public and is therefore the fissionable material of concern.

NOTE: Ingrowth of U-236 and Pu-239 are gradual over time.

U-236 has a lower rem/g value vs U-235. U-236 mass is low.

Pu-239 mass and dose are low.

Table 14-31 - Results for Additional Fissionable Materials

VENTED EXPERIMENT							
			Accidental Release			Vent Experiment Release	
Fissionable Material	Mass g	Fission Rate f/s	Reactor Building		Public Areas	Public Areas	10 CFR 37 Fraction
			TEDE rem	Thyroid TODE rem	TEDE rem	TEDE rem	
^{232}U	3.99E-02	7.98E+09	1.75E-02	5.48E-01	2.45E-04	3.00E-03	3.42E-06
^{233}U	4.17E-03	8.19E+09	1.80E-02	5.60E-01	2.53E-04	3.00E-03	2.80E-06
^{234}U	1.02E+01	9.20E+09	2.08E-02	6.48E-01	2.83E-04	3.00E-03	2.34E-06
^{235}U	5.21E-03	1.01E+10	2.00E-02	6.14E-01	2.78E-04	3.00E-03	2.30E-06
^{236}U	3.23E+00	1.07E+10	2.13E-02	6.54E-01	2.92E-04	3.00E-03	1.58E-06
^{238}U	5.89E+01	1.35E+10	2.87E-02	8.86E-01	3.76E-04	3.00E-03	1.12E-06
^{237}Np	1.10E+01	1.17E+10	2.62E-02	8.12E-01	3.45E-04	3.00E-03	2.88E-06
^{238}Np	2.72E-03	1.38E+10	2.95E-02	9.13E-01	3.85E-04	3.00E-03	1.77E-06
^{238}Pu	6.92E-01	1.44E+10	3.33E-02	1.04E+00	4.23E-04	3.00E-03	4.50E-06
^{239}Pu	5.84E-03	1.45E+10	3.22E-02	1.00E+00	4.12E-04	3.00E-03	3.91E-06
^{240}Pu	5.41E+00	1.45E+10	3.24E-02	1.00E+00	4.16E-04	3.00E-03	3.55E-06
^{241}Pu	2.06E-02	1.64E+10	3.49E-02	1.08E+00	4.45E-04	3.00E-03	2.74E-06
^{242}Pu	2.02E+01	1.74E+10	3.54E-02	1.10E+00	4.52E-04	3.00E-03	3.02E-06

Table 14-31 – Continued

ENCAPSULATED EXPERIMENT						
			Accidental Release			
Fissionable Material	Mass g	Fission Rate f/s	Reactor Building		Public Areas	10 CFR 37 Fraction
			TEDE rem	Thyroid TODE rem	TEDE rem	
^{232}U	4.88E-01	9.77E+10	2.15E-01	6.71E+00	3.00E-03	3.20E-04
^{233}U	1.62E-01	9.59E+10	2.16E-01	6.70E+00	3.00E-03	2.54E-04
^{234}U	1.08E+02	9.75E+10	2.21E-01	6.88E+00	3.00E-03	1.89E-04
^{235}U	5.61E-02	1.09E+11	2.15E-01	6.61E+00	3.00E-03	1.89E-04
^{236}U	3.32E+01	1.10E+11	2.18E-01	6.71E+00	3.00E-03	1.24E-04
^{238}U	4.70E+02	1.07E+11	2.29E-01	7.07E+00	3.00E-03	6.82E-05
^{237}Np	9.58E+01	1.02E+11	2.27E-01	7.06E+00	3.00E-03	1.91E-04
^{238}Np	2.13E-02	1.08E+11	2.30E-01	7.12E+00	3.00E-03	1.05E-04
^{238}Pu	4.90E+00	1.02E+11	2.36E-01	7.38E+00	3.00E-03	2.44E-04
^{239}Pu	4.00E-02	9.97E+10	2.36E-01	7.37E+00	3.00E-03	2.05E-04
^{240}Pu	3.91E+01	1.05E+11	2.34E-01	7.25E+00	3.00E-03	1.96E-04
^{241}Pu	1.39E-01	1.10E+11	2.35E-01	7.30E+00	3.00E-03	1.41E-04
^{242}Pu	1.34E+02	1.16E+11	2.35E-01	7.27E+00	3.00E-03	1.53E-04

Table 14-32 - Comparison of Results for Mixtures of Fissionable Materials

VENTED EXPERIMENT					
Fissionable Material	**f/s per g**	**Public rem per g**	**Public rem per f/s**	**Restrictive Material for Mixture**	**Notes**
^{232}U	2.00E+11	7.52E-02	3.76E-13	U-233	U-233 rem/g is higher than U-232. U-232 mass is usually low vs U-233.
^{233}U	1.96E+12	7.19E-01	3.66E-13	U-233	
^{234}U	9.03E+08	2.95E-04	3.26E-13	U-235	U-234 mass and rem/g is lower than U-235. U-234 dose is low.
^{235}U	1.94E+12	5.76E-01	2.97E-13	U-235	
^{236}U	3.32E+09	9.28E-04	2.79E-13	U-235	U-235 f/s limit is lower. U-236 mass is low
^{238}U	2.28E+08	5.09E-05	2.23E-13	U-235	U-235 f/s limit is lower. Pu-239 mass from irradiation of U-238 is low.
^{237}Np	1.07E+09	2.72E-04	2.55E-13	Np-237	
^{238}Np	5.08E+12	1.10E+00	2.17E-13	Np-237	Np-237 has lower f/s limit
^{238}Pu	2.09E+10	4.34E-03	2.08E-13	Pu-239	Pu-239 has lower f/s limit and higher rem/g values
^{239}Pu	2.48E+12	5.14E-01	2.07E-13	Pu-239	
^{240}Pu	2.68E+09	5.54E-04	2.07E-13	Pu-239	Pu-239 has lower f/s limit and higher rem/g values
^{241}Pu	7.95E+11	1.46E-01	1.83E-13	Pu-239	Pu-239 has lower f/s limit and higher rem/g values
^{242}Pu	8.65E+08	1.49E-04	1.72E-13	Pu-239	Pu-239 has lower f/s limit and higher rem/g values

Table 14-32 – Continued

ENCAPSULATED EXPERIMENT

Fissionable Material	f/s per g	Public rem per g	Public rem per f/s	Restrictive Material for Mixture	Notes
^{232}U	2.00E+11	6.15E-03	3.07E-14	U-233	U-233 rem/g is higher than U-232. U-232 mass is usually low vs U-233.
^{233}U	1.96E+12	6.06E-02	3.09E-14	U-233	
^{234}U	9.03E+08	2.78E-05	3.08E-14	U-235	U-234 mass and rem/g is lower than U-235. U-234 dose is low.
^{235}U	1.94E+12	5.35E-02	2.76E-14	U-235	
^{236}U	3.32E+09	9.04E-05	2.72E-14	U-235	U-235 f/s limit is lower. U-236 mass is low
^{238}U	2.28E+08	6.38E-06	2.79E-14	U-235	U-235 f/s limit is lower. Pu-239 mass from irradiation of U-238 is low.
^{237}Np	1.07E+09	3.13E-05	2.94E-14	Np-237	
^{238}Np	5.08E+12	1.41E-01	2.78E-14	Np-237	Np-237 has lower f/s limit
^{238}Pu	2.09E+10	6.12E-04	2.93E-14	Pu-239	Pu-239 has lower f/s limit and higher rem/g values
^{239}Pu	2.48E+12	7.50E-02	3.02E-14	Pu-239	
^{240}Pu	2.68E+09	7.68E-05	2.87E-14	Pu-239	Pu-239 has lower f/s limit and higher rem/g values
^{241}Pu	7.95E+11	2.16E-02	2.72E-14	Pu-239	Pu-239 has lower f/s limit and higher rem/g values
^{242}Pu	8.65E+08	2.24E-05	2.59E-14	Pu-239	Pu-239 has lower f/s limit and higher rem/g values

e.g.U-235 vented experiments: 1.01×10^{10} f/s per 5.21×10^{-3} g = 1.94×10^{12} f/s per g

3.00×10^{-3} rem / 5.21×10^{-3} g = 5.76×10^{-1} rem per g

2.97×10^{-13} rem per f/s = 5.76×10^{-1} rem per g / 1.94×10^{12} f/s per g

Table 14-33 - Examples for Other Experiment Conditions for U-235

VENTED EXPERIMENT																
				Accidental Release		Vented Release										Vented Exp
Material	Mass	Fission Rate	Irradiation Time	Reactor Building		Public Areas	Public Areas	10 CFR 37 Fraction	Flow Rate	Decay Time	Fluence Rates		Stack Gas Setpt	Stack Part Setpt	Exhaust Setpt
					Thyroid							Non-			
	g	f/s	hours	TEDE rem	TODE rem	TEDE rem	TEDE rem		lpm	min	Thermal	thermal	µCi/ml	µCi/ml	µCi/ml
^{235}U	3.41E-03	1.01E+10	2240	2.04E-02	6.27E-01	2.83E-04	3.00E-03	2.38E-06	3	30	1.00E+12	1.00E+12	1.31E-04	9.79E-06	6.67E-01
^{235}U	9.14E-04	1.77E+09	2240	3.51E-03	1.08E-01	4.89E-05	3.00E-03	4.04E-07	12	7.5	1.00E+12	3.00E+11	1.05E-04	8.55E-06	6.14E-01
^{235}U	5.54E-02	1.07E+11	2240	2.12E-01	6.53E+00	2.96E-03	3.00E-03	2.44E-05	0.5	180	1.00E+12	3.00E+11	2.36E-04	7.51E-06	8.34E-01
^{235}U	7.90E-04	1.53E+09	2240	3.03E-03	9.31E-02	4.22E-05	3.00E-03	3.49E-07	3	6.67	1.00E+12	3.00E+11	1.03E-04	8.38E-06	6.11E-01
^{235}U	1.26E-02	2.45E+10	2240	4.84E-02	1.49E+00	6.75E-04	3.00E-03	5.58E-06	3	60	1.00E+12	3.00E+11	1.53E-04	9.43E-06	7.12E-01
^{235}U	6.66E-01	1.01E+10	2240	1.96E-02	6.03E-01	2.75E-04	3.00E-03	2.24E-06	3	30	1.00E+10	1.00E+08	1.29E-04	9.91E-06	6.66E-01
^{235}U	2.26E-03	1.01E+10	2240	2.01E-02	6.18E-01	2.80E-04	2.99E-03	2.33E-06	3	30	2.00E+12	1.00E+12	1.30E-04	9.83E-06	6.67E-01

Table 14-33 – Continued

ENCAPSULATED EXPERIMENT												
				Accidental Release							Stack Gas Setpt	Stack Part Setpt
Fissionable	Mass	Fission Rate	Irradiation Time	Reactor Building		Public Areas	10 CFR 37 Fraction	Fluence Rates				
					Thyroid TODE				Non-			
Material	g	f/s	hours	TEDE rem	rem	TEDE rem		Thermal	thermal		µCi/ml	µCi/ml
^{235}U	2.43E-02	1.08E+11	17520	2.16E-01	6.64E+00	3.01E-03	1.91E-04	2.00E+12	1.00E+12		1.27E-03	9.30E-03
^{235}U	7.27E+00	1.10E+11	17520	2.14E-01	6.58E+00	3.00E-03	1.86E-04	1.00E+10	1.00E+08		1.29E-03	9.57E-03
^{235}U	7.87E-03	1.09E+11	17520	2.15E-01	6.60E+00	3.00E-03	1.88E-04	8.00E+12	1.30E+12		1.28E-03	9.46E-03
^{235}U	3.93E-03	1.09E+11	17520	2.15E-01	6.59E+00	3.00E-03	1.88E-04	1.60E+13	2.60E+12		1.28E-03	9.46E-03

Notes: Public TEDE is 0.003 rem in all cases
TEDE in reactor building is < 1 rem in all cases
Thyroid TODE in reactor building is < 10 rem in all cases

Data for Other Fissionable Materials

Supporting data used in these calculations for other fissionable materials is given in Table 14-34 through Table 14-37.

Table 14-34 – Fission Cross Sections (barns) for Other Fissionable Materials

Nuclide	Thermal Fission Cross Section (b)	Non-thermal Fission Cross section (b)
^{232}U	7.71E+01	4.16E+02
^{233}U	5.31E+02	7.62E+02
^{234}U	6.70E-02	1.17E+00
^{236}U	6.13E-02	4.34E+00
^{237}U	1.70E+00	4.44E+01
^{238}U	2.65E-05	3.01E-01
^{238}Np	2.03E+03	2.01E+03
^{238}Pu	1.79E+01	2.75E+01
^{240}Pu	5.92E-02	3.36E+00
^{241}Pu	1.01E+03	1.06E+03
^{242}Pu	2.56E-03	1.15E+00

Table 14-35 – Cumulative Fission Yields per 100 Fissions and Fission Cross Sections (barns)

Nuclide	^{235}U		^{233}U		^{239}Pu		^{237}Np	
	Thermal	Non-thermal	Thermal	Non-thermal	Thermal	Non-thermal	Thermal	Non-thermal
	per 100 fissions		per 100 fissions		per 100 fissions		per 100 fissions	
^{83m}Kr	0.536	0.575	0.989	0.964	0.297	0.315	0.342	0.482
^{85m}Kr	1.290	1.358	1.603	1.463	0.563	0.594	1.000	0.689
^{85}Kr	0.283	0.296	1.699	1.519	0.123	0.138	0.203	0.698
^{87}Kr	2.560	2.539	3.921	3.777	0.989	1.037	1.850	1.700
^{88}Kr	3.550	3.433	5.119	4.846	1.272	1.288	2.230	2.080
^{89}Kr	4.510	3.974	5.266	4.925	1.453	1.450	2.610	2.150
^{131m}Xe	0.041	0.035	0.039	0.041	0.042	0.043	0.044	3.600
^{133m}Xe	0.189	0.197	0.168	0.171	0.231	0.245	0.183	6.470
^{133}Xe	6.700	6.710	5.955	6.044	7.016	6.970	6.480	6.470
^{135m}Xe	1.100	1.264	0.830	0.863	1.837	2.083	1.560	7.110
^{135}Xe	6.540	6.579	6.255	6.403	7.605	7.539	7.680	7.250
^{137}Xe	6.130	5.979	6.030	6.220	6.008	5.578	4.030	6.350
^{138}Xe	6.300	5.996	4.896	5.323	5.169	4.710	5.740	5.290
^{131}I	2.890	3.218	3.611	3.746	3.856	3.878	3.160	3.600
^{132}I	4.310	4.661	4.953	5.035	5.389	5.316	4.570	4.850
^{133}I	6.700	6.704	5.955	6.044	6.972	6.908	6.470	6.460
^{134}I	7.830	7.628	6.127	6.048	7.406	7.114	7.310	6.950
^{135}I	6.280	6.274	5.028	5.228	6.539	6.079	6.900	6.720
^{83}Br	0.540	0.576	0.989	0.964	0.297	0.315	0.342	0.482
^{84}Br	0.967	1.011	1.635	1.596	0.429	0.463	0.487	0.706
^{90}Sr	5.873	5.598	6.829	6.429	2.155	2.071	3.525	3.415
^{137}Cs	3.250	3.758	5.191	5.349	4.277	4.660	2.322	3.899
^{147}Pm	2.247	2.143	1.737	1.683	2.007	1.992	2.502	2.225
Cross Section (barns)	**585.0**	**571.0**	**531.0**	**762.0**	**748.0**	**789.0**	**0.020**	**1.330**

Table 14-36 – Cumulative Thermal Fission Yields per 100 Fissions for Other Fissionable Materials

NUCLIDE	^{232}U	^{233}U	^{240}Pu	^{242}Pu
^{83m}Kr	1.48E+00	9.89E-01	2.28E-01	1.10E-01
^{85m}Kr	1.96E+00	1.60E+00	3.25E-01	2.57E-01
^{85}Kr	2.19E+00	1.70E+00	3.28E-01	2.57E-01
^{87}Kr	4.13E+00	3.92E+00	8.15E-01	6.65E-01
^{88}Kr	5.54E+00	5.12E+00	1.17E+00	8.35E-01
^{89}Kr	5.09E+00	5.27E+00	1.24E+00	9.01E-01
^{131m}Xe	4.09E-02	3.94E-02	3.64E-02	2.85E-02
^{133m}Xe	1.62E-01	1.68E-01	1.91E-01	1.67E-01
^{133}Xe	5.81E+00	5.96E+00	6.71E+00	5.84E+00
^{135m}Xe	5.81E-01	8.30E-01	1.11E+00	1.22E+00
^{135}Xe	5.24E+00	6.25E+00	7.24E+00	7.52E+00
^{137}Xe	6.31E+00	6.03E+00	6.22E+00	5.91E+00
^{138}Xe	3.84E+00	4.90E+00	5.69E+00	6.20E+00

^{131}I	3.75E+00	3.61E+00	3.34E+00	2.61E+00
^{132}I	5.15E+00	4.95E+00	4.82E+00	3.97E+00
^{133}I	5.81E+00	5.96E+00	6.71E+00	5.84E+00
^{134}I	4.37E+00	6.13E+00	7.53E+00	7.37E+00
^{135}I	3.52E+00	5.03E+00	6.74E+00	7.42E+00
^{83}Br	1.48E+00	9.89E-01	2.28E-01	1.10E-01
^{84}Br	1.74E+00	1.64E+00	3.94E-01	2.34E-01
^{90}Sr	6.62E+00	6.83E+00	1.87E+00	1.27E+00
^{137}Cs	6.73E+00	5.19E+00	3.72E+00	1.98E+00
^{147}Pm	1.21E+00	1.74E+00	2.12E+00	2.38E+00

Table 14-37 – Cumulative Non-Thermal Fission Yields per 100 Fissions for Other Fissionable Materials

NUCLIDE	^{234}U	^{236}U	^{237}U	^{238}Np	^{238}Pu	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
^{83m}Kr	1.16E+00	5.00E-01	3.77E-01	3.55E-01	3.61E-01	2.75E-01	3.01E-01	2.06E-01	1.75E-01	1.42E-01
^{85m}Kr	1.49E+00	9.57E-01	7.08E-01	5.11E-01	4.95E-01	6.91E-01	4.26E-01	3.02E-01	3.21E-01	2.56E-01
^{85}Kr	1.53E+00	9.62E-01	7.09E-01	5.12E-01	5.08E-01	6.91E-01	4.36E-01	3.05E-01	3.22E-01	2.56E-01
^{87}Kr	2.94E+00	2.24E+00	1.71E+00	1.40E+00	1.16E+00	1.59E+00	1.02E+00	8.40E-01	7.81E-01	6.15E-01
^{88}Kr	3.80E+00	2.78E+00	2.61E+00	1.73E+00	1.53E+00	1.81E+00	1.27E+00	9.33E-01	9.49E-01	7.84E-01
^{89}Kr	4.79E+00	3.77E+00	3.39E+00	2.20E+00	1.82E+00	2.60E+00	1.47E+00	1.32E+00	1.24E+00	1.07E+00
^{131m}Xe	4.07E-02	3.22E-02	3.50E-02	3.79E-02	4.27E-02	3.59E-02	4.23E-02	3.84E-02	3.66E-02	3.39E-02
^{133m}Xe	1.84E-01	1.96E-01	1.56E-01	1.64E-01	1.67E-01	1.92E-01	1.97E-01	1.97E-01	1.84E-01	1.89E-01
^{133}Xe	6.46E+00	6.88E+00	5.46E+00	5.77E+00	5.87E+00	6.72E+00	6.97E+00	6.92E+00	6.46E+00	6.64E+00
^{135m}Xe	8.08E-01	9.88E-01	1.09E+00	1.05E+00	9.48E-01	1.15E+00	1.00E+00	1.14E+00	1.13E+00	1.13E+00
^{135}Xe	5.57E+00	6.09E+00	6.65E+00	6.44E+00	6.74E+00	7.02E+00	7.54E+00	7.45E+00	7.08E+00	7.02E+00
^{137}Xe	5.56E+00	5.34E+00	5.06E+00	6.10E+00	5.96E+00	4.61E+00	5.55E+00	6.22E+00	5.93E+00	5.76E+00
^{138}Xe	5.78E+00	6.27E+00	5.80E+00	5.87E+00	4.99E+00	5.16E+00	4.72E+00	5.67E+00	5.72E+00	5.76E+00
^{131}I	3.74E+00	2.95E+00	3.21E+00	3.48E+00	3.92E+00	3.29E+00	3.88E+00	3.52E+00	3.36E+00	3.11E+00
^{132}I	4.36E+00	4.39E+00	4.77E+00	5.15E+00	5.30E+00	5.15E+00	5.33E+00	5.03E+00	4.71E+00	4.49E+00
^{133}I	6.46E+00	6.88E+00	5.46E+00	5.77E+00	5.87E+00	6.72E+00	6.97E+00	6.92E+00	6.46E+00	6.64E+00
^{134}I	5.62E+00	7.68E+00	6.72E+00	7.98E+00	6.39E+00	7.45E+00	7.13E+00	7.38E+00	7.47E+00	7.34E+00
^{135}I	4.89E+00	5.98E+00	6.62E+00	6.34E+00	5.74E+00	6.99E+00	6.08E+00	6.90E+00	6.83E+00	6.87E+00
^{83}Br	1.16E+00	5.00E-01	3.77E-01	3.55E-01	3.61E-01	2.75E-01	3.01E-01	2.06E-01	1.75E-01	1.42E-01
^{84}Br	1.90E+00	9.39E-01	8.05E-01	6.27E-01	5.89E-01	6.25E-01	4.77E-01	3.61E-01	3.13E-01	2.92E-01
^{90}Sr	6.09E+00	4.83E+00	4.02E+00	2.75E+00	2.47E+00	3.36E+00	2.07E+00	1.83E+00	1.59E+00	1.40E+00
^{137}Cs	3.78E+00	2.06E+00	1.27E+00	1.89E+00	5.04E+00	1.09E+00	4.66E+00	3.97E+00	2.66E+00	2.77E+00
^{147}Pm	2.02E+00	2.30E+00	2.61E+00	2.39E+00	2.23E+00	2.59E+00	1.99E+00	2.20E+00	2.22E+00	2.37E+00

15. DISCUSSION OF CALCULATION UNCERTAINTIES

Dose estimates are made in this analysis and are used to establish experiment controls and radiation monitor setpoints. Dose assessment and compliance with TS requirements is made using radiation monitoring, air sampling, and environmental surveillance data. Uncertainties associated in estimating radiation dose from the assumptions and models used in the calculations and dose assessment from monitoring and environmental surveillance are discussed below.

Analysis Specific Uncertainties:

- Saturation activity of fission gases and halogens are assumed at all times. Saturation would occur for a few, but not all, of the fission gases and halogens under typical irradiation conditions; e.g. 8 hours irradiation followed by 16 hours of decay. For the case of non-stop, continuous irradiation the saturation activity assumption is valid.
- Cumulative fission yields were used to calculate saturation activities. The cumulative yield accounts for the decay of precursors, but does not account for other nuclear reactions that may occur (e.g. capture). Use of cumulative fission yields gives a higher saturation activity.
- Minor differences are reported for fission cross-section and cumulative fission yield data. Fluence rate measurement differences are also minor.
- Complete and instantaneous release of fission gases and halogens from the sample is assumed. This assumption is very conservative, and may be incorrect depending on the sample material and irradiation apparatus/equipment used. This assumption is made since the release fraction cannot be predicted.
- Worst case weather conditions with full occupancy time by members of the public is assumed. Fumigation weather conditions at a wind speed of 1 m/s were used in the radiation dose analysis. From Section 14 Calculation 5, the maximum [X/Q] value under fumigation conditions is conservative compared to the Gaussian Plume Model (GPM) at a wind speed of 1 m/s by a factor of approximately 4 for a 24 hour period [8.54×10^{-3}s/m^3 vs 2.15×10^{-3}s/m^3], and the maximum [X/Q] value under fumigation conditions is conservative compared to calm winds at a wind speed of 0.5 m/s by a factor of approximately 20 for a 24 hour period [8.54×10^{-3}s/m^3 vs 3.99×10^{-4}s/m^3]. Actual weather conditions at the time of release are not monitored, and therefore the worst case weather is assumed even though such conditions may be infrequent.
- No credit for use of the R-63 fan exhaust for the reactor stack is taken in the dose estimates since these fans are not part of the reactor facility and are not required for reactor operation. Operation of the R-63 fan is monitored. If the R-63 fan is operable, the effective stack height is increased which gives a lower [X/Q] value and a lower radiation dose. The effective stack height ranges from 32 m to 70 m depending on wind speed. [X/Q] value and dose are reduced by a factor of 2.2 for fumigation conditions at 1 m/s and by a factor of 1.7 for sector averaged weather for times > 96 h for the GPM.[4]

General Modeling Uncertainties:

- Uncertainties in dose assessments made on the basis of monitoring results incorporate both uncertainties in monitoring data and uncertainties in dosimetric models. The largest uncertainty is usually that associated with the modelling performed using source monitoring data as the input because in this case the modelling includes the dispersion of radionuclides in the environment, which can be predicted only with significant uncertainty. The uncertainties in dose assessments are lower when data from comprehensive environmental monitoring are used and lowest when individual monitoring data are available.[37]

- Besides the uncertainties associated with monitoring procedures, an important source of uncertainty arises from the modelling and especially from people's habits. Often only nationwide average values, if any at all, for the relevant parameters are known, which may deviate considerably from the values for specific persons in specific areas.[37]
- In the case of long-term public exposure with slowly changing radiation conditions, dose assessments should be based on the available data from environmental monitoring in combination with simple static or equilibrium models. Dose assessment can be assigned an estimated uncertainty that takes into account the uncertainties in the parameters of the dosimetric models.[37]
- As stated in reference 37; Both uncertainties in monitoring data and major sources of uncertainties in dosimetric models as presented in Table 15-1 for dosimetry and air sampling should be taken into account in determining the uncertainties in dose assessments made on the basis of monitoring results.

Table 15-1 – Major Sources of Uncertainties in Dose Estimations Made on the Basis of Data from Environmental and Individual Monitoring.

Pathway of Human Exposure	Quantity Monitored	Source of Uncertainties in Dose Estimates
External exposure	Gamma dose monitored by dosimeter	• Location and duration of stay by people • Location of dosimeter relative to occupied areas
Inhalation	Activity concentration in air	• Location of people relative to air sampling • Dose coefficients

- Mathematical modeling for pathway analysis of radiation doses to members of the public caused by radioactive materials in the environment may be complex to meet the challenges encountered. However, the rule of thumb is that the simplest model that will adequately address the situation always should be applied first.[38] Simple models often are highly conservative, but they rely on fewer data than complex models. Initial assessments should be conducted with very simple models; more detailed models and more detailed assessments should be made as data and knowledge of the system being modeled improve.[38]
- Misinterpretation of modeling results can occur when inappropriate boundary conditions or assumptions have been used. The results of any modeling application should be viewed as estimates of reality, and not reality itself. In many cases, seemingly minor changes in assumptions or input can cause drastic changes in the results obtained.[38]
- In many situations, site-specific data are not available, so default parameters or datasets can be used in the transport calculations. These default values often are obtained from generic datasets and are designed to give conservative dose overestimates.[38]

Additionally, the following observations are noted:

- TS radiation dose is prospectively analyzed and experimental conditions are set to meet TS requirements. It is important to note that compliance with the TS radiation dose criteria is made using radiation monitor and air sample data. Radiation monitor calibration error is 10 percent and volume and flow rate measurement error is 10 percent, giving a combined error of 14 percent.
- The TEDE of 0.003 rem in publicly occupied areas outside the reactor building is not capable of being accurately and conclusively monitored with radiation dosimetry given that background radiation level, which are on the order of 10 to 15 µrem/h, or 0.09 to 0.13 rem per year. Therefore, estimates of radiation dose are routinely made from the radiation monitoring system data and air sample analysis.
- Radiation monitors have a response time of a few minutes and alarm within seconds if the setpoint is reached. The radiation monitor setpoints are based on a release at the concentrations that give a TEDE of 0.003 rem over the release period. Given the rapid response time as compared to the release time, there is sufficient time to take action to mitigate or halt the release.
- Radiation dose from airborne activity releases are analyzed from monitor data monthly and after abnormal alarms using facility procedures. If the TEDE of 0.003 rem is being approached, there is sufficient time to take action to mitigate or halt the release.

Summary of calculation uncertainties

It is noted that conservative assumptions were made in this analysis; (1) saturation activity of fission gases and halogens are assumed at all times, (2) complete and instantaneous release of fission gases and halogens, and (3) worst case weather conditions with full occupancy time by members of the public. For typical irradiation conditions and types of fissionable materials used in fueled experiments, these conservative assumptions over-estimate radiation dose. These assumptions provide a significant margin of safety in the estimated radiation dose.

Radiation dose estimates as calculated are conservative by a factor of 2 with the most likely value being 0.003 rem or less. This is mostly due to the assumptions made about the source term being at saturation activities and weather conditions (e.g. wind speed, fumigation vs. GPM). An overestimate by a factor of 2 may also apply to dose assessment if credit is taken for the R-63 fan being operable.

Radiation dose assessment based on monitoring and air sampling data, other uncertainties in instrumentation, detection (counting), and sampling (e.g. volume or flow rate, filter retention) need to be accounted for using accepted error analysis methods. Combined uncertainties for these items are in the range of 15 percent. Accepted atmospheric models and dose conversion factors from accepted dosimetric models were used in the analysis and are recognized to have uncertainties. In lieu of individual or site specific data, use of such data is acceptable for both dose estimation and dose assessment.

16. CONCLUSIONS

Radiation dose from the release of fission gases and halogens are used as TS limits for fueled experiments. The TS limits are well below regulatory limits and provide a consistent basis for fueled experiments. This approach allows fueled experiments to be performed using any fissionable material under various experimental conditions that meet TS radiation dose criteria.

All fueled experiments are limited to a radiation dose of three percent of the annual radiation dose limit for members of the public outside the reactor building; i.e. a TEDE of 0.003 rem in occupied areas outside the reactor building.

TS require an experiment specific analysis to be performed and reviewed for fueled experiments. Utilizing the analysis methodology as detailed, the calculated fission rate for the experiment irradiation conditions is used as an experimental parameter to meet TS radiation dose criteria. For mixtures, the fission rate limit used is for the fissionable material that produces most of the public radiation dose. In this analysis, equations have been given that were utilized for the calculation of the radiation dose from an accidental encapsulation failure and planned release of fission gases and halogens inside and outside the reactor building. Radiation dose from the release of fission gases and halogens is controlled by limiting the fission rate of the fueled experiment. Radiation doses from released radionuclides vary due to differences in the fission yields, fission cross-sections, the thermal to non-thermal fluence rate ratio, and release conditions.

In this analysis, equations have been given for the calculation of radiation monitor setpoints that meet the radiation dose criteria in the proposed TS. Fueled experiments are monitored for radioactivity at the reactor exhaust stack. Additionally, vented fueled experiments are monitored at the experiment exhaust before being connected to the reactor building ventilation system. Control room annunciation occurs for abnormal releases and the confinement system is activated if an alarm setpoint is reached.

In this analysis, equations have been given for calculation of the activity of Sr-90, Cs-137, and Pm-147. The radiation dose of a TEDE of 0.003 rem outside the reactor building limits the fission rate, which depends on the sample mass and fluence rate. At the fission rate limit for the fluence rates achievable in the experimental facilities, the activity produced are several orders of magnitude below 10 CFR Part 37 limits.

In all experimental cases analyzed, it was verified that a TEDE of 0.003 to publicly occupied areas outside the reactor building is met. The TEDE of 0.003 rem is a factor of 3.3 times lower than the constraint radiation dose given in 10 CFR Part 20 (TEDE of 0.01 rem), and is 33 times lower than the annual radiation dose limit for members of the public given in 10 CFR Part 20 (TEDE of 0.1 rem), i.e. only 3 percent. Given the low radiation dose and the conservatisms applied as detailed in this analysis, any dose to the public from fueled experiments performed at the facility would be minimal and indiscernible from background levels.

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