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18 April 2022

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC

Re: License Amendment Request for Fueled Experiments
License No. R-120
Docket No. 50-297

Please find a license amendment request (LAR) for fueled experiments attached. This request is based on revised analyses and replaces all previous submittals. NC State desires this amendment request supporting fueled experiments to be expedited to permit current experimental projects to move forward. We are therefore requesting to separate it from the license renewal process. We will supplement the license renewal application to remove the fueled experiment analysis content once this LAR is processed approved.

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

I declare under penalty of perjury that the forgoing is true and correct. Executed on 18 April 2022.

Sincerely,

Ayman Hawari

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

Enclosures: Attachment 1: Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor
Attachment 2: Technical Specification Changes for Fueled Experiments LAR
Attachment 3: Technical Specifications Amendment 19

ATTACHMENT 1:
Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor

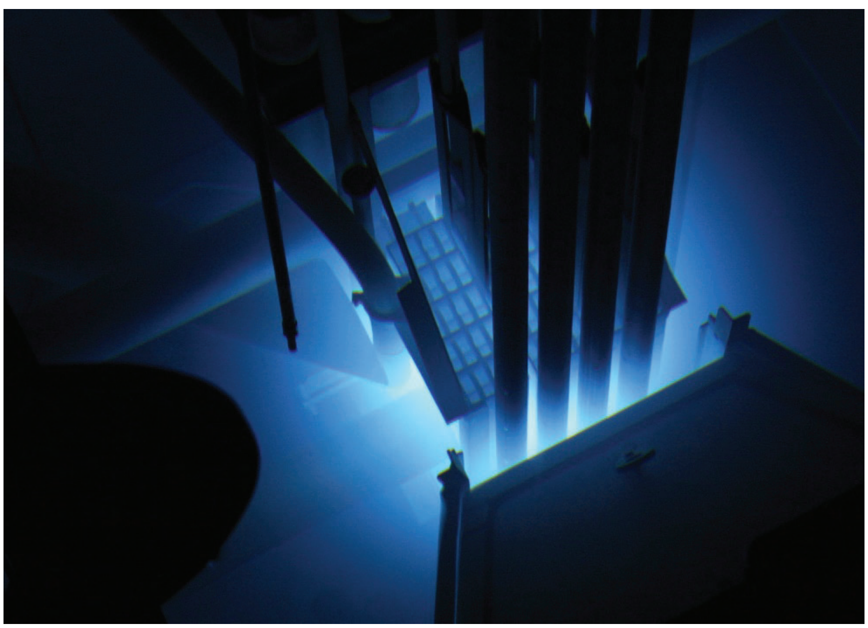
NC STATE UNIVERSITY

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

Nuclear Reactor Program

NORTH CAROLINA STATE UNIVERSITY

RALEIGH, NORTH CAROLINA 27695



LICENSE NO. R-120

DOCKET NO. 50-297

25 March 2022

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EXECUTIVE SUMMARY

This license amendment request (LAR) to enhance the ability of the NC State PULSTAR Facility to perform fueled experiments has been updated to incorporate additional technical specification (TS) requirements. The additional proposed TS provide robust limits against which to verify compliance and adherence to the primary safety basis for fueled experiments. **TS limits are based on a TEDE of 0.01 rem for vented experiment releases and accidental releases from an encapsulated experiment per calendar year from fission gases and halogens in occupied areas outside the reactor building.** All planned airborne releases are monitored and controlled to meet the TEDE dose constraint of 0.01 rem given in 10 CFR Part 20 for airborne effluent. Accidental releases are not expected. The revised TS, associated bases, and supporting calculations are presented in the updated sections of the analyses and discussions presented herein.

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 solid fissionable material (e.g. LEU) under active irradiation and high temperature conditions. Another vented experiment allowing for the measurement of fission product release rates from a fueled molten salt mixture is planned for installation in an in-pool facility. Encapsulated fueled experiments would involve the sealing of fissionable materials in suitable containers and irradiation to required fluences, and are classified as either encapsulated, or encapsulated and submerged. The accidental activity release limits for the encapsulated and submerged experiment are higher since credit is taken for the significant retention of halogens and particulates in the primary coolant. 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 providing robust TS limits against which to verify compliance and adherence to a strong safety basis.

The primary safety basis proposed for all fueled experiments is to limit the maximum permissible radiation dose to less than or equal to a Total Effective Dose Equivalent (TEDE) of 0.01 rem per calendar year in occupied areas outside the reactor building from an accidental or planned release of fission gases and halogens.

To assure compliance with this primary basis, TS limits have been developed for both encapsulated experiments and vented experiments.

- TS requirements for encapsulated fueled experiments give maximum activity releases of I-131 in the event of the failure of either a submerged or non-submerged experiment encapsulation. Submerged experiment encapsulation is irradiated in contact with and fully submerged by primary coolant, so credit is taken for halogen retention in water in the event of encapsulation failure.

- TS requirements for vented fueled experiments include a maximum activity release rate, a minimum experiment exhaust decay time, minimum experiment exhaust filter efficiency, and the operability of experiment exhaust gas flow and radiation monitors with setpoints assuring that a) an alert annunciation is provided to operators if the experimental activity average release rate approaches TS limits, and b) the experiment is automatically isolated prior to TS release rate limits being exceeded. Setpoints for the facility stack gas, stack exhaust (auxiliary), and stack particulate monitors are adjusted to assure that a) an alert annunciation is provided to operators if the experimental activity average release rate approaches TS limits, and b) the facility ventilation system is automatically placed in confinement if elevated concentrations of fission gases occur from a fueled experiment accident.

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 0.01 rem as discussed above for all operational and credible failure modes. 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 assure that the TEDE limits are 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 the TEDE limits of 0.010 rem for members of the public in occupied areas outside the reactor building is maintained by controlling the activity released.

The TS basis for a TEDE of 0.01 rem to members of the public from airborne activity releases for a planned vented experiment **is the radiation dose constraint given in 10 CFR Part 20. The dose constraint of 0.01 rem is 10 times lower than the annual radiation dose limit for members of the public given in 10 CFR Part 20 (TEDE of 0.1 rem)**. Given the low radiation dose and the conservatisms applied as detailed in this analysis, radiation dose to the public from vented fueled experiments performed at the facility would be minimal compared to background levels.

The TS basis for a TEDE of 0.010 rem for an accidental release from a vented or an encapsulated fueled experiment is **equal to the radiation dose constraint given in 10 CFR Part 20 (TEDE of 0.010 rem), and 10 times lower than the annual radiation dose limit for members of the public given in 10 CFR Part 20 (TEDE of 0.1 rem)**. Accidental releases result in maximum radiation doses below twenty percent of the annual occupational TEDE dose limit and fifty percent of the total organ dose-equivalent limit given in 10 CFR Part 20 inside the reactor building.

Accidental releases from vented and encapsulated fueled experiments are 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.

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) Experiment conditions are demonstrated limiting the release of fission gases and halogens produced in a fueled experiment in a calendar year that would result in a TEDE of less than or equal to 0.010 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 1.2.9.e 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 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 1.2.9.e – 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 fission rate. The fission rate 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 1.2.9.e is revised to:

e. **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 flow rate monitor is added as a requirement to verify compliance with TS 3.8.d.iv.(5).
- The vented fueled experiment exhaust gas radiation monitor is added as a requirement to verify compliance with TS 3.8.d.iv.(6). 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.

Alarm setpoints for the reactor stack radiation monitors were changed based on a release from a vented fueled experiment accident that does not exceed a public TEDE of 0.01 rem or an Emergency Action Level given in the Emergency Plan. Alert setpoints are set for abnormal or accidental releases. Alert setpoints are lower than alarm setpoints. The vented fueled experiment exhaust radiation monitor isolates the experiment exhaust and provides control room annunciation if the release exceeds TS limits. The vented fueled experiment exhaust gas flow rate monitor automatically isolates the experiment exhaust and provides control room annunciation if the gas flow rate exceeds the setpoint required to meet the minimum delay time TS. These setpoints were chosen to avoid the unintended initiation of the confinement system from a planned and analyzed vented fueled experiment while ensuring TS limitations are met.

Details on the new setpoints are provided in Section 14 Calculation 10. The bases of TS 3.5 were 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 vented fueled experiment accident. Due to detector response and use of the confinement system the TEDE to the public would be less than 0.01 rem from a vented fueled experiment accident release.

Technical Specification 3.5 is revised in its entirety to include:

3.5 Radiation Monitoring Equipment

Applicability

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective

To assure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specification

The reactor shall not be operated nor shall irradiated fuel or irradiated fueled experiments be moved within the reactor building unless the radiation monitoring equipment listed below and in Table 3.5-1 are operable.(1)(2)(3)(7)

a. Three fixed area monitors operating in the Reactor Building with their setpoints as listed in Table 3.5-1.(1)(3)(4)

b. Stack Gas and Stack Particulate building exhaust monitors continuously sampling air in the facility exhaust stack with their setpoints as listed in Table 3.5-1.(1)(3)(4)

c. The Radiation Rack Recorder.[5]

Table 3.5-1: Required Radiation Area Monitors		
Monitor	Alert Setpoint	Alarm Setpoint
Control Room	≤ 2 mR/hr	≤ 5 mR/hr
Over-the-Pool	≤ 5 mR/hr	≤ 100 mR/hr
West Wall	≤ 5 mR/hr	≤ 100 mR/hr
Stack Gas	Sum of count rate from 1.1×10^{-5} µCi/ml of Ar-41 and 1.3×10^{-4} µCi/ml of Fission Gas [6][9]	Lower count rate from 9.4×10^{-5} µCi/ml of Ar-41 or 5.4×10^{-4} µCi/ml of Fission Gas [6][9]
Stack Particulate	Lower count rate from 4.4×10^{-7} µCi/ml of Co-60 or 1.0×10^{-6} µCi/ml of Fission Halogens [6][9]	Lower count rate from 6.6×10^{-7} µCi/ml of Co-60 or 1.7×10^{-6} µCi/ml of Fission Halogens [6][9]

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

e. Vented fueled experiment flow rate monitor continuously monitoring the experiment exhaust gas flow.[7][8]

(1) For periods of time, not to exceed ninety days, for maintenance to the radiation monitoring channel, the intent of this specification will be satisfied if one of the installed channels is replaced with a gamma-sensitive instrument which has its own alarm audible or observable in the control room. Refer to SAR Section 5.

(2) The Over-the-Pool Monitor may be bypassed for less than two minutes during return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

(3) Stack Gas and Particulate are based on the AEC quantities present in the ventilation flow stream as it exits the stack. Refer to SAR Section 10 for setpoint bases for the radiation monitoring equipment.

(4) May be bypassed for less than one minute immediately after starting the pneumatic blower system.

(5) During repair and/or maintenance of the recorder not to exceed 90 days, the specified area and effluent monitor readings shall be recorded manually at a nominal interval of 30 minutes when the reactor is not shutdown. Refer to SAR Section 5.

(6) Co-60 and Ar-41 are radionuclides of concern without a fueled experiment in progress.

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

[8] Vented fueled experiment exhaust gas radiation and flow monitor setpoints meet Specification 3.8.d.iv and isolate the experiment exhaust when exceeded.

[9] Fission gases and halogens are released from vented fueled experiments or fueled experiment accidents.

Bases

A continued evaluation of the radiation levels within the Reactor Building will be made to assure the safety of personnel. This is accomplished by the area monitoring system of the type described in Section 5 of the Safety Analysis Report (SAR).

Evaluation of the continued discharge air to the environment will be made using the information recorded from the stack particulate and stack gas monitors.

When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, the building will be automatically placed in confinement as described in SAR Section 5.

To prevent unnecessary initiation of the evacuation confinement system during the return of a pneumatic capsule from the core to the unloading station or during removal of experiments from the reactor pool, the Over-the-Pool Monitor may be bypassed during the specified time interval. Refer to SAR Section 5.

Stack gas and stack particulate setpoints are based on the Notification of Unusual Event Emergency Action Level (EAL) for Ar-41 and Co-60, respectively, during normal operation with no vented fueled experiments being performed.

Radiation dose for EAL are higher than those for fueled experiments. While fueled experiments are performed, the stack gas and stack particulate monitor setpoints are changed to meet setpoints listed in Table 3.5-1.

In addition, the exhaust fission gases and exhaust flow rate from vented fueled experiments are monitored in accordance with Specification 3.8. Upon reaching an alarm setpoint from the vented fueled experiment exhaust gas radiation or flow monitors, the vented fueled experiment exhaust is automatically isolated. Radiation monitor setpoints are analyzed as described in the documentation presented in the Fueled Experiment Analysis Report for TS Amendment 19.

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 experiment limits based on the dose being well below the annual radiation dose limit given in 10 CFR Part 20 for members of the public outside of the reactor building. Three percent of the annual public dose limit is used for planned releases from vented fueled experiments. Ten percent of the annual public dose limit is used for accidental releases from encapsulated experiments.

The three release scenarios are as follows;

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. Details about vented experiment releases are given in Section 14, Calculations 1, 3 &4.

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. Details about an accidental release from an encapsulation failure are given in Section 14, Calculations 2 and 3.

3) An accidental release from a submerged encapsulation failure which results an instantaneous release of fission gases and halogens into the reactor pool with 3% of halogens and 100% of fission gases subsequently released to the reactor building. This release to the reactor building is subsequently ventilated by the 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. Details about an accidental release from a submerged encapsulation failure are given in Section 14, Calculations 12.

   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.
   - Limiting the release rate for vented fueled experiments and I-131 activity for encapsulated fueled experiments.

- Retention in the pool water of 97% of halogen fission products for encapsulated and submerged fueled experiment accidental releases.

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

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.

Specification

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.iv.1.

d. Fissionable material(s) in a fueled experiment shall meet the following:

    i. Physical form shall be solid, powder, or liquid.

    ii. Limited to any mixture of U, Pu, and Np.

    iii. Encapsulated fueled experiments are limited as follows:

        1. The saturation activity of an encapsulated fueled experiment is limited to 0.290 Ci of I-131 unless the criteria of 3.8(d)(iii)(2) is met.

        2. The saturation activity of an encapsulated fueled experiment with the primary encapsulation fully submerged underwater in the reactor pool is limited to 1.57 Ci of I-131.

    iv. Vented fueled experiments shall meet the following:

        1. Only fission gases and halogen releases are allowed.

        2. Released activity per calendar day from fission gases at a rate that does not exceed[1]:

            i. 9.75 Ci per day from 1 to 93 days in a calendar year as depicted in Figure 3.8-1.

            ii. 910 Ci divided by 93 days to 365 days in a calendar year as depicted in Figure 3.8-1.

3. Experiment exhaust is filtered for particulates and halogens.(2)
4. Experiment exhaust is decayed for a minimum of 100 minutes.
5. Experiment exhaust flow rate is monitored.
6. Experiment exhaust gas is monitored for radioactivity.
7. Halogens in the stack particulate radiation monitoring channel are monitored.

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) The released activity is controlled to ensure that the 10 CFR 20 annual dose constraint for all airborne effluent of 0.01 rem is met.

(2) At flow rates specified for the vented fueled experiment, removal efficiency for the filter train shall be 0.99 or greater. Removal efficiency for each filter in the filter train shall be 0.95 or greater. Filter efficiencies shall be maintained for the entire duration that the filter is in use.

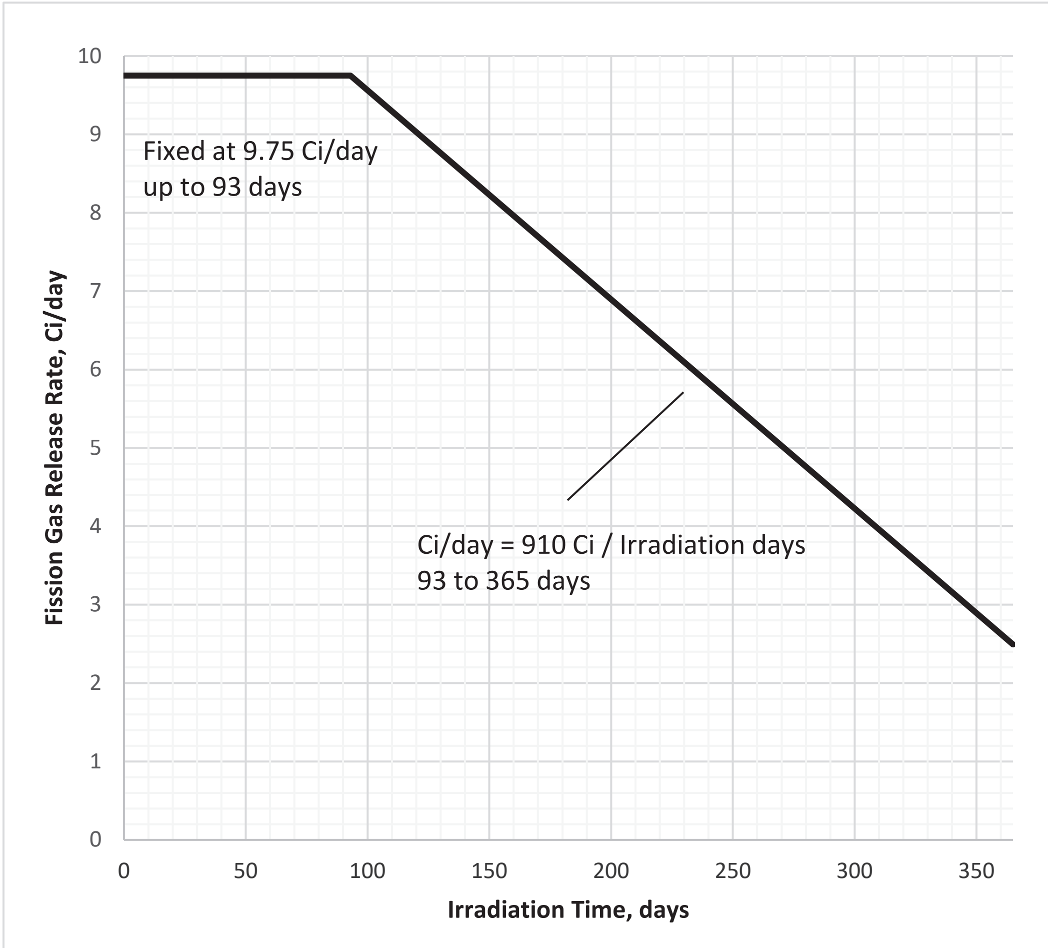


Figure 3.8-1: Vented Fueled Experiment Fission Gas Release Rate

Bases:

1) Specification 3.8.a requires all specifications pertaining to experiment reactivity given in TS 3.2 be satisfied 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 ensuring releases are monitored and filtered if an abnormal release is detected.

3) Specification 3.8.c requires all specifications pertaining to limitations on experiments given in TS 3.7 be satisfied 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 lists the fissionable materials allowed in fueled experiments.

c. Specification 3.8.d.iii gives the saturation activity of I-131 for encapsulated experiments that result in a TEDE of 0.01 rem to public areas outside the reactor building if an accidental release were to occur. An encapsulated experiment that is irradiated fully submerged underwater in the reactor pool takes credit for halogen retention in water in the event of encapsulation failure. Fission gas radionuclides of Kr include 83m, 85m, 85, 87, 88, and 89. Fission gas radionuclides of Xe include 131m, 133m, 133, 135m, 135, 137, and 138. TEDE doses are calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19.

d. Specification 3.8.d.iv limits fission gas release and provides experimental controls that result in a TEDE of 0.01 rem or less in public areas outside the reactor building from vented fueled experiments. Fission gas radionuclides of Kr include 83m, 85m, 85, 87, 88, and 89. Fission gas radionuclides of Xe include 131m, 133m, 133, 135m, 135, 137, and 138. TEDE doses are calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19.

e. For Specifications 3.8 d.iii and 3.8 d.iv, radiation doses were calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19. Footnote (1) to TS 3.8.d.iv.3 specifies the required filter removal efficiency. Meeting Specification 3.8.d.iii and 3.8 d.iv gives a TEDE less than 1 rem and the Total Organ Dose-Equivalent to the thyroid (TODE) less than 25 rem to occupants inside the reactor building.

f. Specification 3.8.e requires all specifications pertaining to criticality control given in TS 5.3 be satisfied thus ensuring that fueled experiments are stored in sub-critical configurations.

g. 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.

TS 4.4 – Radiation Monitoring Equipment

TS 4.4 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 4.4.

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. Filter replacement of 2 years to 2.5 years (30 months), is consistent with TS 4.5 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.

Setpoint verification is changed from weekly to monthly in TS 4.4. The weekly requirement was associated with setpoints that were set with a dial position on an analog meter face and the concern that the dial setting may drift. The radiation monitors now in use have non-volatile memory for the settings. A monthly check is sufficient to verify the settings are not changed. This verification applies to all equipment listed in TS 3.5.

Technical Specification 4.4 is revised in its entirety to:

4.4 Radiation Monitoring Equipment

Applicability

This specification applies to the surveillance requirements for the area and stack effluent radiation monitoring equipment.

Objective

The objective is to assure that the radiation monitoring equipment is operable.

Specification

a. The area and stack monitoring systems shall be calibrated annually but at intervals not to exceed fifteen (15) months.

b. The setpoints shall be verified monthly, but shall not exceed (6) weeks.

c. Vented fueled experiment exhaust gas radiation and flow monitors shall be calibrated prior to initial operation of the experiment and annually, not to exceed 15 months, thereafter for as long as the experiment is in operation.

d. Filter replacements for vented fueled experiments shall be biennial, but shall not exceed 30 months, and shall have a removal efficiency for iodine adsorption of 0.95 or greater at the specified flow rates used.

e. Leak testing of the filter housing(s) used for vented fueled experiments is performed after new filters are installed and prior to initial use; e.g. by pressure testing.

Bases

These systems provide continuous radiation monitoring of the Reactor Building with a check of readings performed prior to and during reactor operations.

The monthly verification of the setpoints in conjunction with the annual calibration is adequate to identify long term variations in the system operating characteristics.

Filter replacement of 2 years to 2.5 years (30 months), is consistent with TS 4.5 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.

FUELED EXPERIMENT ANALYSIS

INTRODUCTION

This analysis supports an amendment to the Technical Specifications (TS) for defining and limiting fueled experiments. The release of fission gases and halogens is used to limit fueled experiments in TS. Three release scenarios are evaluated; (1) experiments with a planned vented release, (2) accidental releases from a dry environment for a vented experiment or for a fully encapsulated experiment, and (3) accidental releases from a fully encapsulated experiment that is completely submerged in the reactor pool.

TS limits each fueled experiment based on the Total Effective Dose Equivalent (TEDE) tor members of the public outside the reactor building from the release of fission gases and halogens in a calendar year. TS limits are based on a TEDE of 0.01 rem from vented experiment releases or accidental releases from any fueled experiment. Given the limits for radiation dose from these releases and conservative assumptions for the release, radiation doses are calculated to be within the TEDE dose constraint of 0.01 rem and well below the 10 CFR Part 20 regulatory limit of 0.1 rem for members of the public outside the reactor building. This approach allows fueled experiments to be performed using fissionable materials under various experimental conditions that meet TS release criteria.

All planned releases are monitored and controlled at the experiment exhaust. Accidental releases from encapsulation failures are not expected. Accidental releases from an encapsulated experiment submerged in the reactor pool takes credit for partial retention of halogens in water. A fueled experiment may extend beyond a calendar year. There may be more than one fueled experiment per calendar year.

For all airborne effluent, 10 CFR Part 20 and TS reporting requirements are followed. Reports are made as required in 10 CFR Part 20 if the TEDE from airborne effluent exceeds 0.01 rem per calendar year and/or 0.1 rem per calendar year. Monitoring, experiment and administrative controls, and frequent dose assessment from airborne effluent is made to ensure the 10 CFR Part 20 dose constraint of 0.01 rem per calendar year is not exceeded for all airborne effluent. Air monitoring and analysis of air samples are performed at frequencies as required by TS and facility procedures which provide sufficient time to halt or alter abnormal releases thereby ensuring compliance with TS and 10 CFR Part 20. Compliance with the TS release criteria is made using radiation monitor and air sample analysis data.

In this analysis, radiation dose calculations from the release of fission gases and halogens to members of the public outside the reactor building are provided for experiment conditions that meet a TEDE of 0.01 rem for a vented experiment and for an encapsulation failure. TS limits the activity released from an accidental encapsulation failure and the activity release rate from a vented fueled experiment. Other experiment conditions necessary for fueled experiments are given as TS requirements. These other requirements include filtration, experiment flow rates, radiation monitoring, and minimum decay time. Saturation activity of fission gases and halogens are used for the released activity in the analysis. 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. Equations and detailed calculations are provided in the analysis. 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 calculations, 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 (1.2.9.e).

Methodology for the Review of Proposed Fueled Experiments

When fueled experiments are proposed, the methodology used in this analysis will be applied to review experimental parameters for compliance with TS and for determining setpoints for related instrumentation. Compliance with TS requirements is analyzed and confirmed for each fueled experiment prior to being performed. Each new fueled experiment and any changes to previously approved fueled experiments will be analyzed before the experiment is conducted. Analysis of a proposed fueled experiment are performed as described in this report and documented to meet TS review requirements.

The activity release rate is determined for vented fueled experiments such that the annual dose constraint of 0.01 rem will not be exceeded. The activity release rate depends on various experimental parameters, such as; experiment volume, experiment exhaust flow rate, experiment delay volume, and experiment exhaust radiation monitor setpoint.

All fissionable materials that produce activity over the course of the experiment irradiation are to be included and evaluated in the analysis of proposed fueled experiments. I-131 saturation activity is determined for encapsulated experiments. For vented and encapsulated experiments, an accidental release analysis is performed.

The experiment analysis will also verify that the TEDE is less than 1 rem and thyroid TODE is less than 25 rem inside the Reactor Building for accidental releases.

Additionally, radiation safety items are analyzed for fueled experiments; e.g. handling, storage, and disposition of the sample and mitigation and control of personnel radiation dose, airborne activity release, and contamination. Other items are analyzed for the proposed fueled experiment; e.g. reactivity, security, emergency response, instrumentation, and compliance with the reactor license and TS.

All new and changed fueled experiments are considered untried and require approval as specified in TS 6.2.3 and TS 6.5. Various documentation is reviewed and approved prior to performing each fueled experiment; e.g. supporting calculations, radiation dose analysis, compliance with fueled experiment possession limits on the reactor license and TS limits, 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. Individual halogen filters have a minimum removal efficiency of 90 percent. A filter train with multiple filters may be used in series to ensure a halogen removal efficiency of 0.99 is maintained while in use.
3. A continuous, controlled release rate from the reactor building ventilation system during the experiment irradiation time is assumed. A minimum decay time of 100 minutes is used.
4. Release and radiation dose is assessed for periods of 2 hours, 24 hours, 96 hours, and 96 to 8760 hours.
5. A constant neutron fluence rate over the entire mass of the fissionable material is 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 the reactor experimental facility used.
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, and encapsulated and submerged, fueled experiments:

9. Radioactive materials are encapsulated until the time of failure.
10. Single-mode nonviolent failure of the encapsulated experiment releases and uniformly distributes radioactive noble gases and halogens into the minimum free air volume of the reactor building.
11. Single-mode nonviolent failure of the encapsulated and submerged experiment releases radioactive noble gases and halogens into the reactor pool.
12. Retention in the pool water of 97% of halogen fission products for encapsulated and submerged fueled experiment accidental releases.

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

13. A constant neutron fluence rate over the entire mass of the fissionable material is 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.

14. 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.
15. Radioactive noble gases and halogens are assumed to be present at the saturation activity from irradiation at the maximum fluence rate for the reactor experimental facility used.
16. 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.
17. 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.
18. An accidental release from a vented or encapsulated experiment is assumed to occur instantaneously and to be uniformly distributed into minimum free air volume of the reactor building.
19. An accidental release from an encapsulated and submerged experiment releases radioactive noble gases and halogens into the reactor pool.
20. 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.
21. 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.
22. Confinement filter removal efficiencies, or retention, are 99.97 percent for particulates with a diameter of 0.3 microns or larger and 90 percent or greater for halogens.
23. 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.
24. Reactor shut-down is assumed during an accidental release.

Fueled Experiment Definition

Conditions for fueled experiment definition:

25. 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.
26. 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

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

or $k = 2.703\times10^{-29}$ 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 rates [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 a dry irradiation facility located either outside or inside the reactor pool. Release of fission gases is not affected by the presence of water.

Measured fluence rates measured at 1 MW operation for reactor experimental facilities were used in this analysis to determine the saturation activity of the fission gases and halogens that are assumed to be released. A thermal neutron fluence rate of $1\times10^{12}\ cm^{-2}s^{-1}$ and a non-thermal fluence rate of $3\times10^{11}\ cm^{-2}s^{-1}$ were used in this analysis.

If lower fluence rates are used, the mass of fissionable materials used may be increased so that the same release occurs. If higher fluence rates are used, the mass of fissionable materials used is decreased so that the same release occurs. Experimental facilities include beamtubes and dry tubes or standpipes.

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 96 to 8760 hours. The exposure is presumed to last during the irradiation time. Since irradiation times may be divided, full time 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.

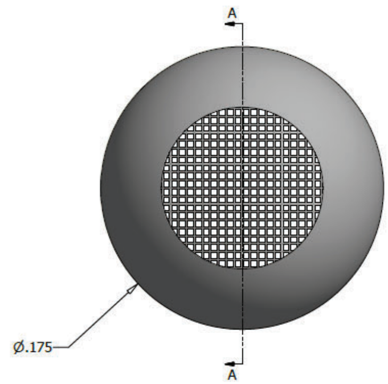
Samples from vented fueled experiments are contained to prevent the release of sample material and allow the release fission gases and halogen vapors. Samples for vented experiments are contained using a sample holder, which is placed in the experimental facility. Examples of sample holders as shown 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

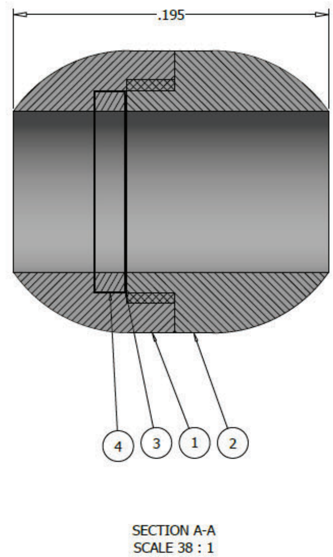
If high, unexpected activity is present in the vented fueled experiment exhaust, then the release is automatically isolated by exceeding the set point of the vented fueled experiment exhaust radiation monitor. In addition, the flow rate of the vented fueled experiment is monitored and is automatically isolated if the flow rate exceeds that which would meet the minimum decay time specification.

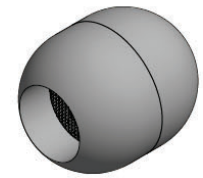
Also, the vented experiment is halted if air sampling indicates an abnormal release is occurring or has occurred following facility health physics procedures.

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

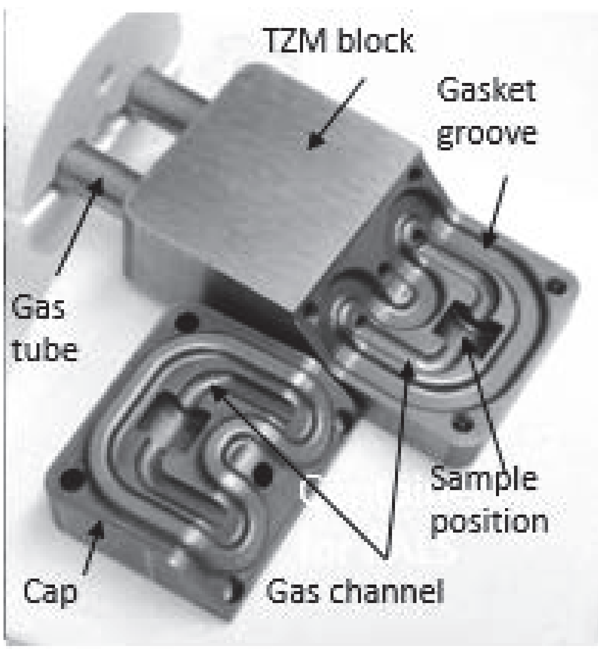


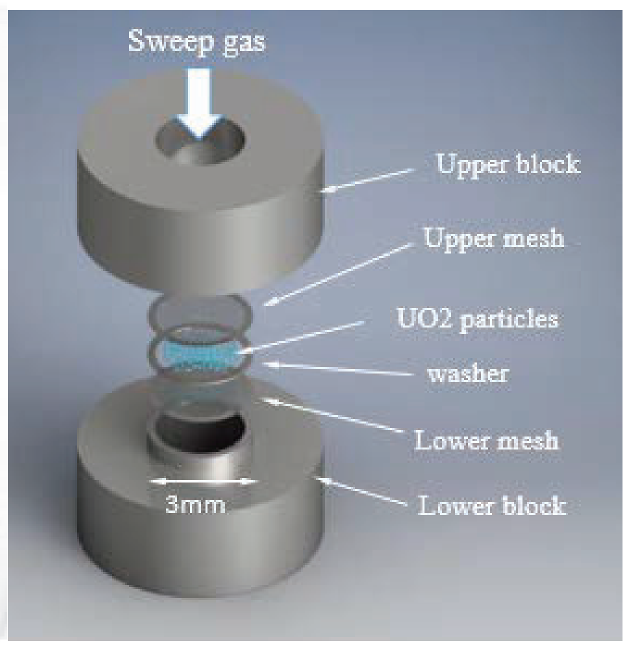
Sample Capsule with Mesh Filter





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 4.5 on the ventilation system, including filter testing in accordance with TS 4.5.e 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

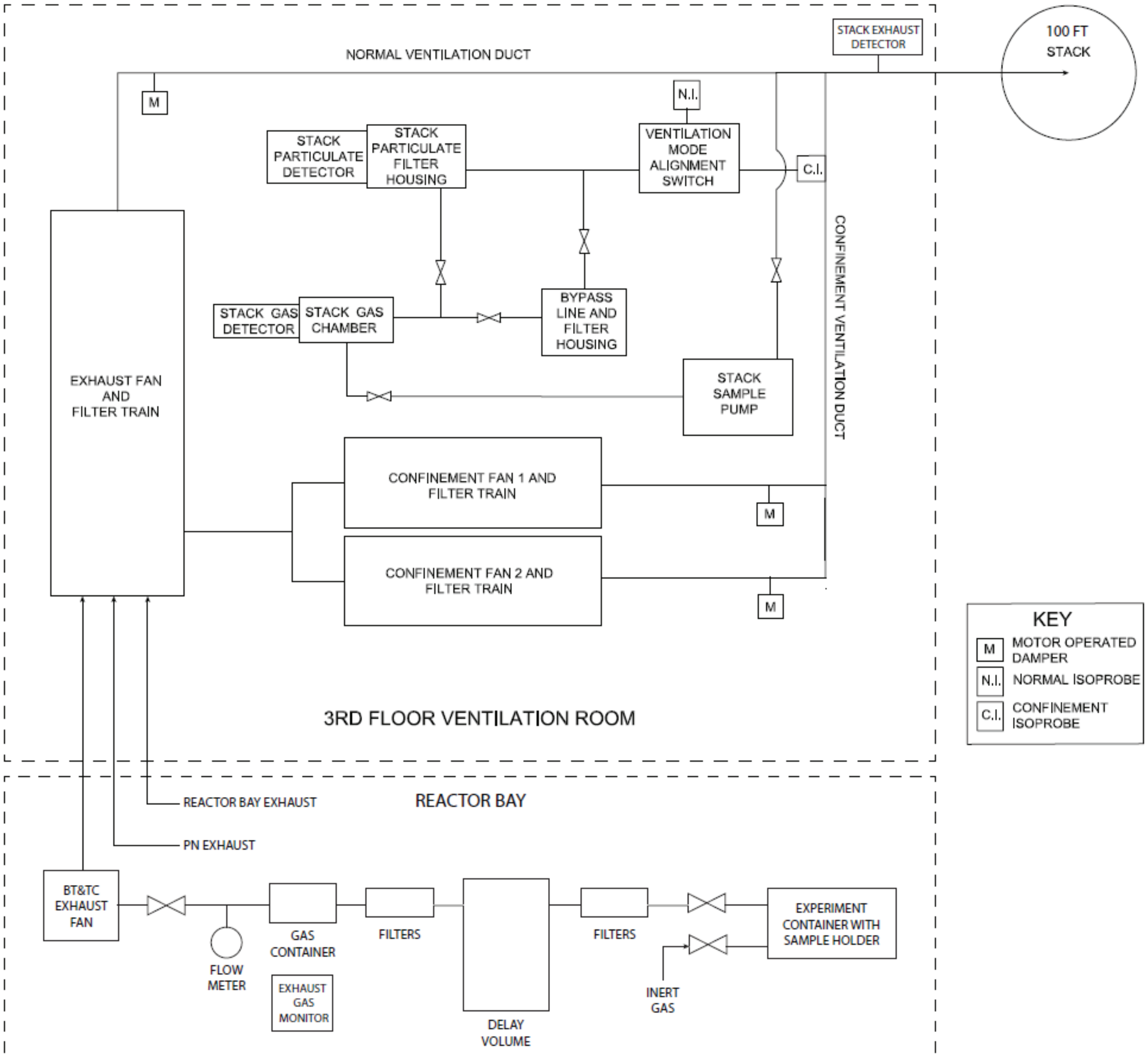
The exhaust from a vented fueled experiment is filtered for halogens and particulates prior to entering the reactor building ventilation system. A generic experiment arrangement is shown in Figure 4-2.

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 4.4 for the vented fueled experiment. The replacement time of 2 years, up to 30 months, is consistent with TS 4.5 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 99.97 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 setpoint at the specified experiment exhaust flow rate assuring that the minimum decay time TS is not exceeded. 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 9 and Section 14 Calculation 10 of this analysis to meet TS limits. TS 3.5 is revised to include the vented fueled experiment exhaust radiation 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, isolation of the vented fueled experiment exhaust and control room annunciation occurs if a) the vented fueled experiment exhaust radiation monitor exceeds the alarm setpoint or if it fails, or b) the vented fuel experiment exhaust flow monitor exceeds the alarm setpoint or if it fails. The vented fueled experiment exhaust radiation monitor provides early detection and control of airborne releases since it is located prior to the reactor building ventilation exhaust and the fission gases are at higher concentrations. Refer to Section 9 for a discussion on alarm setpoints.

Due to the possibility of an abnormal release of radioactive gases or radioiodine being present in airborne effluent, initiation of the confinement system occurs as follows for the stack radiation monitors:

- Stack gas radiation monitor or stack exhaust radiation monitor exceeding an alarm set point. Refer to Section 9 for a discussion on alarm setpoints.
- 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). Refer to Section 9 for a discussion on alarm setpoints.
- Operator action, as deemed necessary, is needed to initiate the confinement system. 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 (auxiliary)) provide control room annunciation. Continued operation with a radiation monitor annunciation is not a normal condition and would be immediately investigated.
- If any of the stack radiation monitors fail, the confinement system is automatically 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. 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, 39 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 volume is greater than 2.4×10^3 m^3 given in the FSAR and TS. 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 from either a vented or encapsulated fueled experiment, 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. No decay or filtration prior to an accidental release is assumed for vented or encapsulated fueled experiments.

For accidental releases from an encapsulated and submerged fueled experiment, the failure releases radioactive noble gases and halogens into the reactor pool. The pool water retains 97% of halogen fission products and all particulates.

The released materials are then exhausted by the ventilation system and reactor stack to the environment. Concentration inside the reactor building, decay inside the reactor building, halogen and particulate removal by the confinement filters, reactor building exhaust ventilation rate, and atmospheric dispersion are considered in the analysis.

Concentration from an accidental release from a vented, encapsulated, or encapsulated and submerged experiment

The sample is assumed to contain saturated activities of radioactive fission gases (Kr, Xe) and halogens (I, Br) at the time of encapsulation failure. For accidental release from a vented or encapsulated experiment, 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. For accidental releases from an encapsulated and submerged fueled experiment, the failure releases radioactive noble gases and halogens into the reactor pool with 3% of halogens and all of the fission gases released to the reactor building free air space.

Initially, the release occurs for 2 minutes in normal ventilation with no filtration; the confinement system has not been initiated yet. 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:

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

where, C(0) in Ci/m^3 and $A(\infty)$ in Ci

r is retention by reactor pool water; For encapsulated fueled experiments that are submerged in the reactor pool; r = 0.97 for halogens and r = 0 for fission gases. For all other fueled experiments, r = 0 for halogens and fission gases.

V is the minimum reactor building experiment area free air volume = 2.4×10^3 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 in confinement 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 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. Individual halogen filters have a minimum removal efficiency of 90 percent. A filter train with multiple filters may be used in series to ensure a halogen removal efficiency of 0.99 is maintained while in use.
- Each vented fueled experiment has a minimum decay time of 100 minutes. At the maximum experiment exhaust rate, a minimum experiment holdup volume is needed to give the 100 minute minimum 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 automatically and manually 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. Experiment exhaust radiation and flow monitor alarm annunciation is provided locally and in the control room with automatic isolation of the experiment exhaust.

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 prior to filtration

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

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

where, $c(\infty)$ in Ci/m^3

$A(\infty)$ is in Ci

w is the experiment hold up volume

Decay time, t

Filtered concentration from a vented experiment at the reactor stack

The experiment exhaust is filtered before entering the reactor building ventilation system. Two halogen filters are used in the experiment exhaust. Then the experiment exhaust is diluted by the reactor building ventilation system exhaust operating in the normal mode and released to the environment at the reactor stack. Concentration at the reactor stack, C, is given by:

$$C = c(\infty)[p/(p+F)][1\text{-R}]^2 \qquad \text{EQ. 5-5}$$

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

$$C = [A(\infty)/w]e^{(-\lambda t)}[p/(p+F)][1\text{-R}]^2 \qquad \text{EQ. 5-6}$$

where, C and $c(\infty)$ 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

Filter retention, R = 0.9 for halogens and R = 0 for noble gases

Filtered r**elease rate for a 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:

$$q = c(\infty)[1-R]^2 p \qquad \text{EQ. 5-7}$$

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

$$q = [A(\infty)/w]e^{(-\lambda t)}[1-R]^2 p \qquad \text{EQ. 5-8}$$

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

$c(\infty)$ 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 rate 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 $c(\infty)$ is diluted and exhausted by the same flow rate, F; i.e. Q = q.

Vented fueled experiment exhaust gas radiation and flow rate monitoring

The vented fueled experiment exhaust gas is monitored for flow rate and radioactivity. Refer to Section 9 for a discussion on alarm setpoints.

The experiment exhaust flow rate is specified for each planned vented fueled experiment to meet the 100 minute minimum decay time specification given in TS 3.8(d)(iv.)(4). The experiment exhaust flow rate is indicated locally and in the control room. If the setpoint is exceeded, the experiment exhaust is automatically isolated and annunciation occurs in the control room.

A vented fueled experiment exhaust gas radiation monitor is used for measuring radioactivity in the exhaust gas from a vented fueled experiment. Exceeding the setpoint of the vented experiment exhaust gas radiation monitor automatically isolates the exhaust from a vented fueled experiment and annunciates in the control room.

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. Setpoints for the stack radiation monitors allow normal, expected releases from a vented fueled experiment to occur without causing an alarm signal or control room annunciation. Abnormal releases above normal vented fueled experiment operating levels provides control room annunciation. If an accidental release from a vented or encapsulated fueled experiment at the TS limit occurs, the stack radiation monitor(s) alarm and initiate 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 0.01 rem is reached.

The revised TS 4.4 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 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 made monthly using facility procedures which permits early detection of radiation dose before reaching 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 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 maximum calculated atmospheric dispersion {X/Q] value at a given distance is used 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:

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

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):

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

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):

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

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:

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

where, 2.032 = (16 / 2π) [2 /π]^{½} 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 as ranging from 32 to 43 percent in Greensboro, NC. Inversion duration is not reported. 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.

1 m/s was and a duration of 1 day was used for calculating fumigation [X/Q] in this analysis. Weather conditions given in Table 6-2 were to calculate [X/Q]. For periods of 4 days a wind speed less than 4 m/s was used.

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 infrequent and assumed 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] calculations made for calm winds and fumigation for actual occupied locations are more conservative than those made using the GPM.

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 (μ).

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.

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

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:

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

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

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

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 a vented or encapsulated fueled experiment is limited to the following:

- TEDE to occupants inside the reactor building is limited to 1 rem.
- Thyroid TODE to occupants inside the reactor building is limited to 25 rem.
- TEDE for members of the public outside the reactor building is limited to 0.01 rem.

Dose from a planned, vented fueled experiment is limited to the following:

- TEDE for members of the public outside the reactor building is limited to 0.01 rem.

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 2.4×10^9 ml. Dimensions were set at 50 feet high by 41 feet deep and 41.3 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:

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

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

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

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

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

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, 10]

Release Rate and I-131 Activity

The fission gas release rate from vented fueled experiments and I-131 saturation activity from encapsulated fueled experiments are used as limits in TS. For vented fueled experiments, experiment exhaust filtration and decay as stated in the assumptions are used to determine the TS release limit. For accidental releases from encapsulated experiments, fission gases and halogens are released into and exhausted from the reactor building air space with filtration by the confinement system. For accidental releases from encapsulated and submerged experiments, fission gases and halogens are released into the reactor pool with 3% of the halogens and all of the fission gases released into the reactor building air space. Separate TS limits for I-131 saturation activity for encapsulated and encapsulated and submerged fueled experiments are used. I-131 saturation activity is indicative of all fission gases and all halogens released from an encapsulation failure.

At the TS planned release rate limit for vented experiments a TEDE of 0.01 rem to members of the public outside the reactor building is calculated. At the TS release I-131 activity limit for encapsulated experiments, a TEDE of 0.01 rem to members of the public outside the reactor building is calculated from an accidental release. The fission product inventory is less for vented experiments than for encapsulated experiments and therefore accidental releases from vented experiments have a TEDE less than 0.01 rem in public areas outside the reactor building. Additionally, it is calculated that accidental releases in the reactor building air space from encapsulated and vented experiments have a TEDE less than 1 rem and a thyroid TODE less than 25 rem to occupants during the reactor building evacuation time of 6 minutes.

An experiment specific fission rate and mass may be determined before a fueled experiment is performed that meet TS limits. Both of these depend on fueled experiment conditions; fissionable material used, form, neutron fluence rates, and vented fueled experiment decay time.

An experiment fission rate for a given set of conditions is given by:

$$\left(\frac{f}{s}\right) = \left(TEDE\ in\ rem \middle/ public\ TEDE\ rem\ per\ unit\ fission\ rate\right) \qquad \text{EQ. 9-1}$$

where, TEDE to members of the public outside the reactor building is 3×10^{-3} rem for a vented fueled experiment and is 1×10^{-2} rem for an encapsulated fueled experiment.
Public TEDE per unit fission rate is taken from EQ. 8-4 for submersion and inhalation pathways.

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:

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

where, N is converted to mass of the fissionable material.

Releases are monitored and assessed during the irradiation to determine compliance with TS and regulatory limits. Halogens are sampled at the experiment exhaust and reactor stack. Analysis of halogen samples is performed upon filter removal as required by facility procedures. Radiation dose assessment of airborne effluent is made monthly using facility procedures. The monitoring and analysis regime permits early detection of radiation dose before reaching regulatory dose constraint levels and regulatory dose limits.

Radiation Monitor Setpoints

Alarm and alert setpoints for the stack gas, stack exhaust (auxiliary), stack particulate, and vented experiment gas exhaust radiation monitors are provided in TS. 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, alarms occur well before the daily release rate limit given in TS is reached. Setpoints for radiation monitors are determined for fueled experiments as described in Section 14 Calculation 10.

Stack Gas and Stack Exhaust (Auxiliary) Radiation Monitoring Channels

Alarm and alert setpoints are determined for the stack gas and stack exhaust (auxiliary) radiation monitor are based on the more limiting of Emergency Action Levels (EAL) given in the facility Emergency Plan for Ar-41 or accidental releases of fission gases from vented or encapsulated fueled experiments at TS limits. No decay prior to release is assumed for vented or encapsulated fueled experiments. The alarm setpoint is set below the initial concentrations of fission gases released from a vented fueled experiment accident. The alert setpoint is set at the sum of maximum concentration of ^{41}Ar released during normal operations and fission gases released from vented fueled experiments at the TS limit.

At the alarm setpoint, the confinement system initiates. At the alert setpoint, control room annunciation occurs.

Stack Particulate Radiation Monitoring Channel

The stack particulate monitor is equipped with a radioiodine cartridge during fueled experiments as required by the revised TS 3.8.

The stack particulate radiation monitoring alarm set point is based on the more limiting of EAL given in the facility Emergency Plan for Co-60 or accidental releases of halogens from vented or encapsulated fueled experiments at TS limits. No decay prior to release is assumed for vented or encapsulated fueled experiments. The alert setpoint is set below the alarm setpoint and is based on halogens released from vented fueled experiments at TS limits or below the EAL based on Co-60.

At the alarm setpoint, the confinement system initiates. At the alert setpoint, control room annunciation occurs.

Vented Fueled Experiment Exhaust Gas Radiation Monitor

The vented fueled experiment exhaust gas radiation monitor is used to measure concentration of fission gases in the experiment exhaust prior to entering the reactor building exhaust system. Kr and Xe fission gases are readily released and are monitored at the experiment exhaust thereby providing an immediate assessment of released activity. The alarm setpoint is based on fission gases released at TS limits after a decay time of 100 minutes. The alert setpoint is below the alarm setpoint.

At the alarm setpoint, the vented fueled experiment exhaust is halted by isolating the exhaust flow. The alert annunciates in the control room and locally at the experiment. Radiation monitor readings are indicated locally and in the control room.

The revised TS 4.4 includes channel calibration of the vented fueled experiment exhaust radiation monitor.

Vented Fueled Experiment Exhaust Flow Rate

The vented fueled experiment exhaust gas flow rate is monitored. The experiment exhaust flow rate setpoint is specified for each planned vented fueled experiment. The experiment exhaust flow rate is indicated locally and in the control room.

At the alarm setpoint, the experiment exhaust is automatically isolated and control room annunciation occurs. If the experiment exhaust flow rate fails, annunciation in the control room occurs and the experiment is automatically isolated. Flow failure may be due to a block or leak. The revised TS 4.4 includes channel calibration of the vented fueled experiment exhaust flow rate meter to test isolation of the experiment exhaust and control room annunciation.

Other Experiment Limits

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.

No revision to the emergency plan is needed.

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

Production of radionuclides listed in Category 2 Quantities of Concern in 10 CFR Part 37

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:

$$A(T) = k\sigma\phi NY\left[1 - e^{\lambda T}\right] \qquad \text{EQ. 12-1}$$

Irradiation time of 10 years at the fission rate limits based on a TEDE of 0.01 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 10 years, the fraction of 10 CFR Part 37 Category 2 activity limits was less than $1.9\text{x}10^{-2}$. At the radiation dose of 0.01 rem TEDE to the public outside the reactor building, 10 CFR Part 37 limits are not exceeded. Details are given in Section 14 Calculation 6.

Possession Limits

The possession limits are within 10 CFR Part 37 Category 2 limits for the fissionable materials requested and associated fission product inventory. The fraction of 10 CFR Part 37 Category 2 activity limits is well below the limiting value of 1 for extended irradiation times at the experiment fluence rates achievable and mass of fissionable materials used.

Storage and Inventory

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 fission rate in Section 14 Calculation 11.

14. CALCULATIONS AND RESULTS

Calculations 1 through 8 were performed for accidental and planned vented releases of fission gases and halogens from the irradiation of U-235, Pu-239, U-233, and Np-237. Equations and data stated in Sections 1 through 10 were used.

Example calculations for vented experiments were made using an experiment exhaust rate of 5 lpm (83.3 ml/s) and a 100 minute decay time, i.e. a decay volume of 500 liters (100 m x 5 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. For vented experiments, the TEDE radiation dose for accidental releases of 0.01 rem to areas outside the reactor building was limiting. For encapsulated experiments, the saturation activity of I-131 in the sample is used to limit the TEDE from accidental releases in public areas outside the reactor building to 0.01 rem.

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. [X/Q]T was evaluated for vented fueled experiments to determine the activity released over the irradiation time, which gives a release rate. [X/Q]T based on the analysis made in this calculation indicates that the 24 h and 2240 h are equal, 96 h is lowest, and 8760 h is highest. Therefore, TS limits are given for two irradiation times; (1) up to 2240 h and (2) from 2240 h to 8760 h. The experiment parameters affecting the release rate may be adjusted and evaluated as needed to meet release rate and activity limits. 2240 h is approximately a typical full time operating year for the reactor facility (9 h/day x 248 work days).

Results are given in Table 14-6 for TS limits from vented and encapsulated fueled experiments.

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 compare results for additional fissionable materials. Results for vented and encapsulated fueled experiments giving three percent of the annual public dose limits outside the reactor building are given. The same equations, assumptions, and irradiation conditions used in Calculations 1 through 3 were used.

Calculation 12 was performed to determine the activity limit for an encapsulated and submerged experiment.

CALCULATION 1 – Planned Release from a Vented Fueled Experiment

Table 14-1 gives the data and results for vented fueled experiments using U-235 at an experiment exhaust flow rate of 5 lpm and a decay time of 100 minutes (500 liters decay volume), and an exposure time of 24 h. Release rates, public TEDE, mass, and fission rates for fissionable materials are provided in Table 14-2.

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

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	U-235		Target atoms, N	1.89E+20	atoms
Mass	7.36E-02	g	Thermal fission rate	1.10E+11	f/s
Mass Number, A	235	g/mol	Non-thermal fission rate	3.23E+10	f/s
Sigma thermal	585	barns	Total fission rate	1.43E+11	f/s
Sigma non-thermal	571	barns			
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
Vented experiment exhaust	83.3	ml/s	v normal	3.68E-04	1/s
Vented experiment exhaust	5	lpm			
Vented experiment volume	500	liters			
Decay time	100	minutes			
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	2.10E+04
^{85m}Kr	1.61E+04	4.30E-05	1.29E+00	1.36E+00	5.03E+04
^{85}Kr	3.39E+08	2.05E-09	2.83E-01	2.96E-01	1.10E+04
^{87}Kr	4.57E+03	1.52E-04	2.56E+00	2.54E+00	9.85E+04
^{88}Kr	1.02E+04	6.78E-05	3.55E+00	3.43E+00	1.36E+05
^{89}Kr	1.89E+02	3.67E-03	4.51E+00	3.97E+00	1.69E+05
^{131m}Xe	1.03E+06	6.74E-07	4.05E-02	3.54E-02	1.52E+03
^{133m}Xe	1.89E+05	3.66E-06	1.89E-01	1.97E-01	7.36E+03
^{133}Xe	4.53E+05	1.53E-06	6.70E+00	6.71E+00	2.58E+05
^{135m}Xe	9.18E+02	7.55E-04	1.10E+00	1.26E+00	4.38E+04
^{135}Xe	3.28E+04	2.12E-05	6.54E+00	6.58E+00	2.52E+05
^{137}Xe	2.29E+02	3.02E-03	6.13E+00	5.98E+00	2.35E+05
^{138}Xe	8.46E+02	8.19E-04	6.30E+00	6.00E+00	2.40E+05
^{131}I	6.93E+05	1.00E-06	2.89E+00	3.22E+00	1.14E+05
^{132}I	8.26E+03	8.39E-05	4.31E+00	4.66E+00	1.69E+05
^{133}I	7.49E+04	9.26E-06	6.70E+00	6.70E+00	2.58E+05
^{134}I	3.16E+03	2.20E-04	7.83E+00	7.63E+00	3.00E+05
^{135}I	2.37E+04	2.93E-05	6.28E+00	6.27E+00	2.42E+05
^{83}Br	8.64E+03	8.02E-05	5.40E-01	5.76E-01	2.11E+04
^{84}Br	1.91E+03	3.63E-04	9.67E-01	1.01E+00	3.76E+04

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	3.86E-07	1.52E-02	5.85E-09
^{85m}Kr	1.33E-06	1.10E+02	1.46E-04
^{85}Kr	3.77E-07	1.74E+00	6.55E-07
^{87}Kr	1.36E-06	5.25E+02	7.12E-04
^{88}Kr	3.09E-06	1.33E+03	4.12E-03
^{89}Kr	1.60E-15	1.20E+03	1.93E-12
^{131m}Xe	5.16E-08	5.48E+00	2.83E-07
^{133m}Xe	2.46E-07	1.99E+01	4.90E-06
^{133}Xe	8.75E-06	2.25E+01	1.96E-04
^{135m}Xe	1.61E-08	2.79E+02	4.50E-06
^{135}Xe	7.60E-06	1.73E+02	1.32E-03
^{137}Xe	1.06E-13	1.10E+02	1.16E-11
^{138}Xe	6.01E-08	7.10E+02	4.27E-05
^{131}I	3.88E-08	3.97E+04	1.54E-03
^{132}I	3.49E-08	1.95E+03	6.81E-05
^{133}I	8.35E-08	7.41E+03	6.18E-04
^{134}I	2.74E-08	1.89E+03	5.19E-05
^{135}I	6.94E-08	2.54E+03	1.76E-04
^{83}Br	4.46E-09	1.09E+02	4.84E-07
^{84}Br	1.45E-09	1.35E+03	1.97E-06
		TOTAL =	9.00E-03
		Fission Gas Total =	6.54E-03
		Halogen Total =	2.46E-04

Supporting Calculations for U-235:

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

Kr-87: 9.85×10^{4} µCi = $(7.36\times10^{-2}g)(6.022\times10^{23} / 235\ g)(2.703\times10^{-29})$

$\times [(585)(1\times10^{12})(2.56/100)+(571)(3\times10^{11})(2.54/100)]$ or 9.85×10^{-2} 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: $1.36\times10^{-6}\mu Ci\text{-}h/ml=(9.85\times10^{4}\mu Ci/5\times10^{5}ml)(83.3ml/s)e^{[-(1.52E\text{-}4/s)(5E5ml)/83.3ml/s]}[1](8.54\times10^{-3}s/m^{3})(1\times10^{-6}m^{3}/ml)(24\ h)$

I133: $8.35\times10^{-8}\mu Ci\text{-}h/ml=(2.58\times10^{5}\mu Ci/5\times10^{5}ml)(83.3ml/s)(e^{[-(9.26E\text{-}6/s)(5E5ml)/83.3ml/s]})[0.01](8.54\times10^{-3}s/m^{3})(1\times10^{-6}m^{3}/ml)(24\ h)$

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

TEDE is 1.96×10^{-4} rem = $(8.75\times10^{-6}$ µCi-h/ml)(22.5 rem per µCi-h/ml)

DCF: DCF listed were taken as described in Section 8 of the analysis for submersion and inhalation.

Fission rate limit from Eq. 9-1: 1.43×10^{11} f/s = 9.00×10^{-3} rem / (6.31×10^{-14} rem per f/s)

The public dose from 1.43×10^{11} f/s for U-235 is 9.00×10^{-3} rem which gives 6.31×10^{-14} rem per f/s.

Mass of fissionable material from Eq. 9-2: $N = (f/s)/[\sigma\phi]$

1.89×10^{20} atoms = 1.43×10^{11} f/s / $[(585\times10^{-24}cm^{2})(1\times10^{12}cm^{-2}s^{-1})+(571\times10^{-24}cm^{2})(3\times10^{11}cm^{-2}s^{-1})]$

Table 14-2: **Vented Fueled Experiment Release Rates for U-235**

Nuclide	**Eq. 4-8 Vented Exp Release Rate, q Ci/h**
83mKr	6.77E-03
85mKr	2.33E-02
85Kr	6.61E-03
87Kr	2.38E-02
88Kr	5.43E-02
89Kr	2.82E-11
131mXe	9.06E-04
133mXe	4.32E-03
133Xe	1.54E-01
135mXe	2.83E-04
135Xe	1.33E-01
137Xe	1.86E-09
138Xe	1.06E-03
131I	6.82E-04
132I	6.14E-04
133I	1.47E-03
134I	4.82E-04
135I	1.22E-03
83Br	7.83E-05
84Br	2.55E-05
Fission Gas Total =	4.08E-01
Halogen Total =	4.57E-03

Release Rate in Ci/h = q (3600 s/h)(1 Ci/ 1x10^6 µCi) and q is taken from EQ 5-8

$$q = [A(\infty)/w]e^{(-\lambda t)}[1 - R]p$$

where, q is the decayed, filtered release rate in µCi/s that enters the reactor building ventilation system.
The reactor stack release rate, Q, is equal to the experiment release rate, q.
p is the experiment exhaust rate in ml/s
with A(∞) in µCi and w = experiment volume, ml
λ is the radioactive decay constant in 1/s
t is decay time in seconds
Filter retention; R = 0.99 for halogens and R = 0 for noble gases

Examples:

Kr-88: 5.43x10^{-2}Ci/h = (1.36x10^{5}µCi/5x10^{5}ml)(e$^{-6.78E-5/s x 100m x 60s/m}$)(1)(83.3ml/s)(3600s/h)(1Ci/1x10^{6}µCi)

I-133: 1.47x10^{-3}Ci/h = (2.58x10^{5}µCi/5x10^{5}ml)(e$^{-9.26e-6/s x 100m x 60s/m}$)(0.01)(83.3ml/s)(3600s/h)(1Ci/1x10^{6}µCi)

CALCULATION 2: Accidental Release from an Encapsulated Fueled Experiment

Tables 14-3 and 14-4 below give the data and results for accidental releases from encapsulated fueled experiments using Pu-239. For accidental releases, a public TEDE ≤ 0.01 rem and a thyroid TODE < 25 rem and TEDE inside the reactor building is < 1 rem are used as dose limits.

Table 14-3 – Calculation 2 – Accidental Release from Experiment using Pu-239

PARAMETER VALUES					
Parameter	**Value**	**Units**	**Parameter**	**Value**	**Units**
Nuclide	Pu-239		Target atoms, N	2.82E+20	atoms
Mass	1.12E-01	g	Thermal fission rate	2.11E+11	f/s
Mass Number, A	239	g/mol	Non-thermal fission rate	6.68E+10	f/s
Sigma thermal	748	barns	Total fission rate	2.78E+11	f/s
Sigma non-thermal	789	barns			
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
			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					
			Cumulative Yield %		
Nuclide	**Half-Life (sec)**	**Decay Constant (1/s)**	**Thermal Fission**	**Non-Thermal Fission**	**Eq. 2-1 Saturation Activity (µCi)**
^{83m}Kr	6.70E+03	1.04E-04	2.97E-01	3.15E-01	2.26E+04
^{85m}Kr	1.61E+04	4.30E-05	5.63E-01	5.94E-01	4.28E+04
^{85}Kr	3.39E+08	2.05E-09	1.23E-01	1.38E-01	9.50E+03
^{87}Kr	4.57E+03	1.52E-04	9.89E-01	1.04E+00	7.52E+04
^{88}Kr	1.02E+04	6.78E-05	1.27E+00	1.29E+00	9.58E+04
^{89}Kr	1.89E+02	3.67E-03	1.45E+00	1.45E+00	1.09E+05
^{131m}Xe	1.03E+06	6.74E-07	4.24E-02	4.27E-02	3.19E+03
^{133m}Xe	1.89E+05	3.66E-06	2.31E-01	2.45E-01	1.76E+04
^{133}Xe	4.53E+05	1.53E-06	7.02E+00	6.97E+00	5.26E+05
^{135m}Xe	9.18E+02	7.55E-04	1.84E+00	2.08E+00	1.42E+05
^{135}Xe	3.28E+04	2.12E-05	7.60E+00	7.54E+00	5.70E+05
^{137}Xe	2.29E+02	3.02E-03	6.01E+00	5.58E+00	4.43E+05
^{138}Xe	8.46E+02	8.19E-04	5.17E+00	4.71E+00	3.80E+05
^{131}I	6.93E+05	1.00E-06	3.86E+00	3.88E+00	2.90E+05
^{132}I	8.26E+03	8.39E-05	5.39E+00	5.32E+00	4.03E+05
^{133}I	7.49E+04	9.26E-06	6.97E+00	6.91E+00	5.22E+05
^{134}I	3.16E+03	2.20E-04	7.41E+00	7.11E+00	5.51E+05
^{135}I	2.37E+04	2.93E-05	6.54E+00	6.08E+00	4.83E+05
^{83}Br	8.64E+03	8.02E-05	2.97E-01	3.15E-01	2.26E+04
^{84}Br	1.91E+03	3.63E-04	4.29E-01	4.63E-01	3.28E+04

Table 14-3 Continued – Calculation 2 – 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	3.10E-07	2.85E-08	5.94E-07	4.47E-09
^{85m}Kr	5.89E-07	7.44E-08	1.13E-06	8.53E-09
^{85}Kr	1.31E-07	2.25E-08	2.53E-07	1.90E-09
^{87}Kr	1.03E-06	7.79E-08	1.96E-06	1.48E-08
^{88}Kr	1.32E-06	1.44E-07	2.53E-06	1.90E-08
^{89}Kr	1.22E-06	8.05E-09	1.94E-06	1.46E-08
^{131m}Xe	4.40E-08	7.52E-09	8.49E-08	6.39E-10
^{133m}Xe	2.43E-07	4.05E-08	4.68E-07	3.53E-09
^{133}Xe	7.25E-06	1.23E-06	1.40E-05	1.05E-07
^{135m}Xe	1.88E-06	4.56E-08	3.47E-06	2.61E-08
^{135}Xe	7.85E-06	1.15E-06	1.51E-05	1.14E-07
^{137}Xe	5.13E-06	3.95E-08	8.43E-06	6.34E-08
^{138}Xe	4.99E-06	1.13E-07	9.18E-06	6.91E-08
^{131}I	4.00E-06	6.81E-08	7.71E-06	5.80E-08
^{132}I	5.54E-06	5.59E-08	1.06E-05	7.99E-08
^{133}I	7.20E-06	1.15E-07	1.39E-05	1.04E-07
^{134}I	7.50E-06	4.56E-08	1.43E-05	1.07E-07
^{135}I	6.65E-06	9.16E-08	1.28E-05	9.63E-08
^{83}Br	3.10E-07	3.19E-09	5.96E-07	4.48E-09
^{84}Br	4.43E-07	1.91E-09	8.36E-07	6.30E-09

Supporting Calculations:

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

I-131: 2.9×10^{5} µCi = $(1.12\times10^{-1}\text{ g})(6.022\times10^{23}/239\text{ g})(2.703\times10^{-29})$

$\times\,[(748)(1\times10^{12})(3.86/100)+(789)(3\times10^{11})(3.88/100)]$, or 2.9×10^{-1} Ci

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

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

<u>where:</u>

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: Ψr $= (5.26\times10^{5}\ \mu\text{Ci} / 2.4\times10^{9}\text{ ml})(1\text{ h} / 3600\text{ s})$

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

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

$= (7.25\times10^{-5} + 1.4510^{-5})$ µCi-h/ml

$= 2.12\text{x}10^{-5}$ 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

$[X/Q] = 8.54\times10^{-3}\ s/m^3$ for T up to 24 h

$R = 0$

Xe-133: $\Psi p = (5.26\times10^{5}\ \mu Ci / 2.4E9\ ml)(8.54\times10^{-3})(1\ h / 3600\ s)$

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

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

$= (1.46\times10^{-6} + 1.25\times10^{-7})\ \mu Ci\text{-}h/ml = 1.59\times10^{-6}\ \mu Ci\text{-}h/ml$

Table 14-4 – Calculation 2 – 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)	Sub 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	4.70E-10	4.70E-10	4.33E-10	9.01E-10	9.01E-10	6.78E-11
^{85m}Kr			1.10E+02	6.50E-06	6.50E-06	8.20E-06	1.25E-05	1.25E-05	9.41E-07
^{85}Kr			1.74E+00	2.28E-08	2.28E-08	3.92E-08	4.39E-08	4.39E-08	3.31E-09
^{87}Kr			5.25E+02	5.40E-05	5.40E-05	4.09E-05	1.03E-04	1.03E-04	7.76E-06
^{88}Kr			1.33E+03	1.75E-04	1.75E-04	1.92E-04	3.37E-04	3.37E-04	2.53E-05
^{89}Kr			1.20E+03	1.46E-04	1.46E-04	9.66E-06	2.33E-04	2.33E-04	1.75E-05
^{131m}Xe			5.48E+00	2.41E-08	2.41E-08	4.12E-08	4.65E-08	4.65E-08	3.50E-09
^{133m}Xe			1.99E+01	4.84E-07	4.84E-07	8.06E-07	9.32E-07	9.32E-07	7.02E-08
^{133}Xe			2.25E+01	1.63E-05	1.63E-05	2.76E-05	3.14E-05	3.14E-05	2.36E-06
^{135m}Xe			2.79E+02	5.23E-05	5.23E-05	1.27E-05	9.66E-05	9.66E-05	7.27E-06
^{135}Xe			1.73E+02	1.36E-04	1.36E-04	1.98E-04	2.62E-04	2.62E-04	1.97E-05
^{137}Xe			1.10E+02	5.64E-05	5.64E-05	4.34E-06	9.27E-05	9.27E-05	6.98E-06
^{138}Xe			7.10E+02	3.55E-04	3.55E-04	8.05E-05	6.52E-04	6.52E-04	4.91E-05
^{131}I	3.95E+04	1.30E+06	2.42E+02	1.58E-01	5.19E+00	2.71E-03	3.05E-01	1.00E+01	2.31E-03
^{132}I	4.57E+02	7.73E+03	1.49E+03	3.36E-03	4.36E-02	1.09E-04	6.44E-03	8.36E-02	1.56E-04
^{133}I	7.02E+03	2.16E+05	3.92E+02	5.08E-02	1.55E+00	8.50E-04	9.79E-02	2.99E+00	7.74E-04
^{134}I	1.58E+02	1.28E+03	1.73E+03	2.48E-03	1.09E-02	8.62E-05	4.72E-03	2.07E-02	2.03E-04
^{135}I	1.47E+03	3.76E+04	1.06E+03	1.05E-02	2.50E-01	2.32E-04	2.02E-02	4.82E-01	2.44E-04
^{83}Br	1.03E+02		5.09E+00	3.23E-05	1.58E-07	3.46E-07	6.19E-05	3.03E-07	4.87E-07
^{84}Br	1.01E+02		1.25E+03	1.00E-04	5.55E-05	2.58E-06	1.89E-04	1.05E-04	8.53E-06
			TOTAL	2.26E-01	7.05E+00	4.56E-03	4.36E-01	1.36E+01	3.83E-03

Supporting Calculations:

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

Xe-133 dose inside the reactor building:

4.77×10^{-5} rem $= [(7.25\times10^{-6} + 1.40\times10^{-5})$ μCi-h/ml](22.5 rem per μCi-h/ml)(0.1)

$= (1.63\text{x}10^{-5} + 3.14\times10^{-5})$ rem

Xe-133 dose outside the reactor building:

3.00×10^{-5} rem $= [(1.23\times10^{-6} + 1.07\text{x}10^{-7})$ μCi-h/ml]((22.5rem per μCi-h/ml)

$= (2.76\times10^{-5} + 2.36\times10^{-6})$ rem

I-131 dose inside the reactor building:

TEDE is 0.463 rem $= (4.00\times10^{-6} + 7.71\times10^{-6})(3.95\times10^{4}+0.1(242))$

$= (1.58\times10^{-1} + 3.05\times10^{-1})$ rem

Thyroid TODE is 15.2 rem $= (4.00\text{x}10^{-6} + 7.71\times10^{-6})(1.3\times10^{6}+0.1(242))$

$= (5.19 + 10.0)$ rem

I-131 dose outside the reactor building:

TEDE is 5.02×10^{-3} rem $= (6.81\times10^{-8} + 5.80\times10^{-8})(3.95\times10^{4} + 242)$

$= (2.71\times10^{-3} + 2.31\times10^{-3})$ rem

Reactor building TEDE = 0.226 rem + 0.436 rem = 0.662 rem < 1 rem

Reactor building Thyroid TODE = 7.05 rem + 13.6 rem = 20.7 rem < 25 rem

Public TEDE = $4.56\text{x}10^{-3}$ rem + $3.83\text{x}10^{-3}$ rem = $8.39\text{x}10^{-3}$ rem < 0.01 rem

CALCULATION 3: Results for Vented and Encapsulated Fueled Experiments*

Calculations 1 and 2 were performed for U-235, U-233, Pu-239, and Np-237 for vented and encapsulated fueled experiments. Mass and public TEDE dose were determined for each case using the TS limits for release rates and I-131 activity. Results are given in Table 14-5. Encapsulated and submerged fueled experiment results are given in Calculation 12.

In the case of an accidental release, fission gases and halogens are assumed to be released into the reactor building free air volume prior to the decay volume and then removed by the reactor building ventilation system. The confinement system would be activated due to the high concentrations of fission gas in the reactor building exceeding the alarm setpoint of the stack radiation monitors. For this event, reactor shutdown occurs after the confinement system is activated. A release time of 24 hours provides 10 air changes to the reactor building, which would remove essentially all of the airborne activity.

The results from Calculation 2 for an encapsulated fueled experiment accident are scaled for the lower mass for a vented fueled experiment giving a lower TEDE dose for the vented fueled accident.

Table 14-5 – Calculation 3 – Vented and Encapsulated Fueled Experiment Results*

Experiment	Fissionable Material	Mass Limit, g	f/s Limit	Accident RB TEDE rem	Accident RB Thy TODE rem	Accident Public TEDE rem	Vented TEDE rem	Irrad Hours	Exposure Hours	Fission Gas Release Rate Ci/day	Fission Gas Release Ci	I-131 Ci
Vented	U-235	7.36E-02	1.43E+11	2.82E-01	8.68	3.94E-03	9.00E-03	24	24	9.8	9.8	1.1E-01
Encapsulated		1.87E-01	3.63E+11	7.16E-01	22.01	9.98E-03		24	24			2.9E-01
Vented	U-233	6.41E-02	1.26E+11	2.77E-01	8.60	3.88E-03	1.00E-02	24	24	9.8	9.8	1.2E-01
Encapsulated		1.50E-01	3.24E+11	6.47E-01	20.09	9.07E-03		24	24			2.9E-01
Vented	Pu-239	6.38E-02	1.58E+11	3.77E-01	11.76	4.79E-03	7.30E-03	24	24	9.8	9.8	1.7E-01
Encapsulated		1.12E-01	3.31E+11	6.62E-01	20.62	8.40E-03		24	24			2.9E-01
Vented	Np-237	9.09E+01	9.69E+10	2.16E-01	6.69	2.85E-03	5.20E-03	24	24	9.8	9.8	9.4E-02
Encapsulated		2.81E+02	3.40E+11	6.68E-01	20.73	8.81E-03		24	24			2.9E-01

*NOTE: Encapsulated and submerged fueled experiment results are given in Calculation 12.
RB is Reactor Bay.

Supporting calculation:

U-235 TEDE for vented experiment accident = (7.36E-02 g / 0.187 g)(9.98E-03 rem) = 3.94E-3 rem

CALCULATION 4: TS Release Limits

Table 14-2 lists results for all fissionable materials for vented fueled experiments at the TS fission gas release rate limit of 9.75 Ci/day up to 93 days (2240 h). From 93 to 365 days (8760 h), the fission gas release is limited to 910 Ci for vented fueled experiments over the irradiation time, i.e. the release rate is limited to 910 Ci divided by number of days. The release is continuous over the irradiation time.

For vented experiments, A(∞) 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. [X/Q]T was evaluated for vented fueled experiments to determine the activity released over the irradiation time, which gives a release rate. [X/Q]T based on the analysis made in this calculation indicates that the 24 h and 2240 h are equal, 96 h is lowest, and 8760 h is highest. Therefore, TS limits are given for two irradiation times; (1) up to 93 days and (2) from 93 to 365 days. The experiment parameters affecting the release rate may be adjusted and evaluated as needed to meet release rate and release activity limits. 93 continuous days is equal to 2240 h. 2240 h is approximately a typical full time operating year for the reactor facility (9 h/day x 248 work days). Annual dose limits apply to a calendar year by regulation. A year is 8760 h. Vented fueled experiment limiting release rates and released activity for different fissile isotopes resulting in a TEDE of $\leq$0.01 rem to the public are given in Table 14-6.

[X/Q] is 8.54×10^{-3} s/m^3 for 24 h.
[X/Q] is 7.79×10^{-3} s/m^3 for 24 h to 96 h gives a higher release rate. The 24 h release rate limit was used for conservatism.
[X/Q] is 9.15×10^{-5} s/m^3 for times greater than 96 hours.

Table 14-6: **Vented Fueled Experiment Release Rates and Released Activity**

Fissionable Material	Mass, g	f/s	Vented Exp TEDE rem	Fission Gas Ci/day	Fission Gas, Ci	Hours, h
U-235	7.36E-02	1.43E+11	9.00E-03	9.75	9.75	24
U-233	6.41E-02	1.26E+11	1.00E-02	9.75	9.75	24
Pu-239	6.38E-02	1.58E+11	7.30E-03	9.75	9.75	24
Np-237	9.09E+01	9.69E+10	5.20E-03	9.75	9.75	24
U-235	7.36E-02	1.43E+11	9.00E-03	9.75	915.2	2240
U-233	6.41E-02	1.26E+11	1.00E-02	9.75	914.7	2240
Pu-239	6.38E-02	1.58E+11	7.30E-03	9.75	916.9	2240
Np-237	9.09E+01	9.69E+10	5.20E-03	9.75	911.9	2240
U-235	1.88E-02	3.65E+10	9.00E-03	2.5	915.2	8760
U-233	1.64E-02	3.22E+10	1.00E-02	2.5	914.7	8760
Pu-239	1.63E-02	4.05E+10	7.30E-03	2.5	916.9	8760
Np-237	2.32E+01	2.48E+10	5.20E-03	2.5	911.9	8760

Released Fission Gas Activity, Ci = Ci/day x day/24 h x Hours

Encapsulated Fueled Experiment Activity Limit

For encapsulated fueled experiments, the release scenario used in this analysis is an accident with a release averaged over 24 hours resulting in a TEDE of 0.01 rem in public areas outside the reactor building.

Saturation activity is assumed in this analysis at all irradiation times. Accidental releases for U-233, U-235, Pu-239 and Np-237 were analyzed using the calculations, equations, and data provided in this analysis for the experiment fluence rates given in Calculation2 and a 24 hour release.

The saturation activity of I-131 ranges from -10% to +8% of the average for the fissionable materials analyzed, with U-235 being the lowest. Saturation activity of other radionuclides have larger deviations from the average and therefore were not used to set limits on encapsulated fueled experiments.

Results obtained for I-131 are given in Table 14-7.

Table 14-7: Encapsulated Fueled Experiment Saturation Activities

	U-235	U-233	Pu-239	Np-237	Units
131I	0.290	0.320	0.347	0.330	Ci
Average	0.322				Ci
% Difference	9.87	0.54	-7.85	-2.56	-

The minimum saturation activity of 0.29 Ci of I-131 is used as the limit for encapsulated fueled experiments.

Encapsulated and submerged fueled experiment results are given in Calculation 12. I-131 saturation activities for encapsulated and submerged experiments would give the same relative (%) differences.

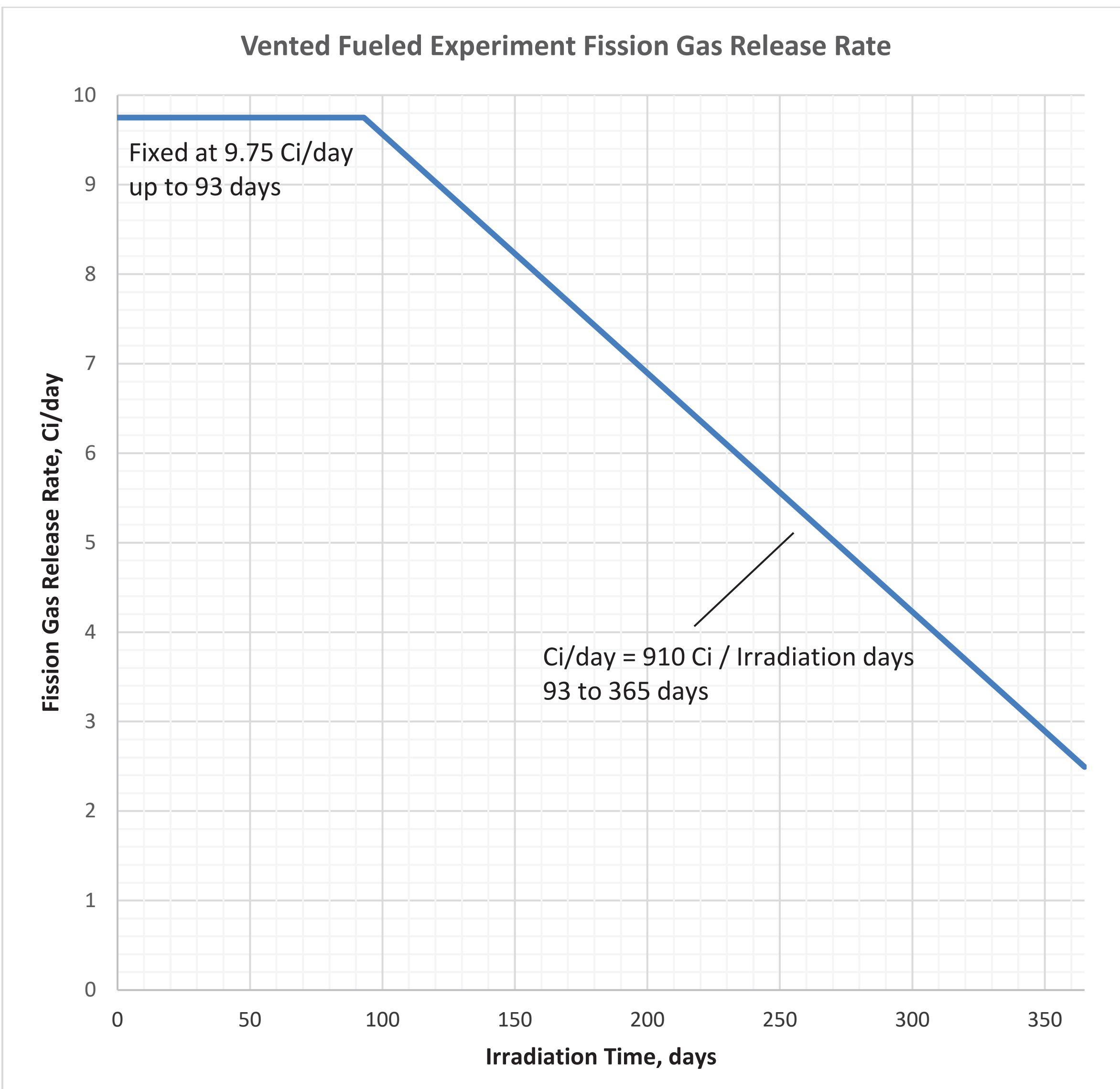
From Calculations 1, 2, and 3, limits used in TS include the following:

1. For vented fueled experiments, results are plotted below. Release rates for vented fueled experiments shall not exceed the line. The release of 9.75 Ci/day is used for irradiation times up to 93 days. A release rate equal to 910 Ci divided by the number of days irradiated is used for times from 93 to 365 days. A graph of release rate for vented experiments is shown in Figure 14-1.

2. TS limits on vented fueled experiments shall include a minimum decay time of 100 minutes.

3. For encapsulated experiments, the activity of I-131 is limited to 0.29 Ci.

Figure 14-1 Vented Fueled Experiment Fission Gas Release Rate



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

Table 14-8 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 14-9 gives the TEDE for other public areas for accidental releases from encapsulation failure and for accidental releases from vented fueled experiments at the TS release limits for U-235.

Table 14-10 gives the TEDE for other public areas for planned releases from vented fueled experiments at the TS release rate limit for U-235.

Table 14-8 – 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] Fumigate 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	4.63E-04	2.31E-04	2.31E-04	1.11E-07	1.51E-06
All	100 to 150	up to 12	2.46E-04	4.78E-04	2.39E-04	2.39E-04	3.63E-06	3.89E-06
All	150 to 5000	up to 30	2.00E-03	4.30E-03	7.57E-03	2.15E-03	7.79E-04	9.15E-05
Withers, Mann	50	12	5.38E-03	1.73E-04	9.73E-05	9.73E-05	3.96E-15	3.71E-09
Broughton, Riddick	70	12	3.97E-03	9.36E-05	1.96E-04	1.96E-04	2.75E-10	2.05E-07
Patterson, Ricks	90	12	3.17E-03	5.80E-05	2.23E-04	2.23E-04	3.14E-08	9.87E-07
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	6.50E-06	4.64E-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	1.39E-05	5.18E-06
North	350	20	8.23E-04	3.99E-06	4.90E-04	1.76E-04	6.00E-05	1.00E-05
MAXIMUM			**8.54E-03**	**4.30E-03**	**7.57E-03**	**2.15E-03**	**7.79E-04**	**9.15E-05**

[X/Q] value analysis notes:

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

Table 14-9 – Calculation 5 - TEDE for other public areas for accidental releases from encapsulation failure and for accidental releases from vented fueled experiments at the TS release limits for U-235

Building or Location	Eq. 8-4 Fumigate TEDE 24 h (rem)	Eq. 8-4 Calm Wind TEDE 24 h (rem)	Eq. 8-4 GPM TEDE 24 h (rem)	Eq. 8-4 Fumigate TEDE 2 h (rem)	Eq. 8-4 Fumigate TEDE 24 h (rem)	Eq. 8-4 Calm Wind TEDE 24 h (rem)	Eq. 8-4 GPM TEDE 24 h (rem)	Eq. 8-4 Fumigate TEDE 2 h (rem)
All	1.0E-02	5.4E-04	2.7E-04	8.0E-03	3.9E-03	2.1E-04	1.1E-04	3.1E-03
All	2.9E-04	5.6E-04	2.8E-04	2.3E-04	1.1E-04	2.2E-04	1.1E-04	9.0E-05
All	3.4E-03	5.0E-03	2.5E-03	2.7E-03	1.3E-03	2.0E-03	9.9E-04	1.1E-03
Withers, Mann	6.3E-03	2.0E-04	1.1E-04	5.0E-03	2.5E-03	8.0E-05	4.5E-05	2.0E-03
Broughton, Riddick	4.6E-03	1.1E-04	2.3E-04	3.7E-03	1.8E-03	4.3E-05	9.0E-05	1.5E-03
Patterson, Ricks	3.7E-03	6.8E-05	2.6E-04	3.0E-03	1.5E-03	2.7E-05	1.0E-04	1.2E-03
DH Hill	2.3E-03	2.5E-05	2.5E-03	1.9E-03	9.2E-04	1.0E-05	9.9E-04	7.4E-04
Cox	2.0E-04	1.9E-05	2.5E-04	1.6E-04	8.0E-05	7.3E-06	1.0E-04	6.4E-05
Dabney	1.8E-03	1.4E-05	6.0E-04	1.4E-03	7.1E-04	5.6E-06	2.4E-04	5.7E-04
Hillsborough St.	1.8E-03	1.4E-05	3.0E-04	1.4E-03	7.1E-04	5.6E-06	1.2E-04	5.7E-04
Talley, Reynolds	1.2E-03	1.4E-05	2.4E-04	9.3E-04	4.6E-04	5.6E-06	9.4E-05	3.7E-04
Carroll, Syme	1.3E-03	5.4E-06	1.9E-04	1.0E-03	4.9E-04	2.1E-06	7.5E-05	3.9E-04
North	9.6E-04	4.7E-06	2.1E-04	7.7E-04	3.8E-04	1.8E-06	8.1E-05	3.0E-04
MAXIMUM	1.0E-02	5.0E-03	2.5E-03	8.0E-03	3.9E-03	2.0E-03	9.9E-04	3.1E-03
	Encap. Failure Accidental Release				**Vented Exp Accidental 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, (DH Hill):

$$2.5\times10^{-3} \text{ rem} = 1\times10^{-2} \text{ rem } (2.17\times10^{-3} \text{ s m}^{-3} / 8.54\times10^{-3} \text{ s m}^{-3})$$

Cox Hall at 24 hours from Calm Wind for a vented experiment accidental release:

$$1.9\times10^{-5} \text{ rem} = 5.0\times10^{-3} \text{ rem } (1.58\times10^{-5} \text{ s m}^{-3} / 4.30\times10^{-3} \text{ s m}^{-3})$$

3.9×10^{-3} rem is the maximum TEDE from a vented fueled experiment accidental release for U-235. The TEDE is proportional to the mass ratio of the vented fueled experiment to the encapsulated experiment;

$$3.9\times10^{-3} \text{ rem} = 0.01 \text{ rem } (7.36\times10^{-2}\text{g} / 0.187\text{g})$$

The vented fueled experiment masses are taken from Calculation 3 Table 14-5.

NOTE: Public dose from encapsulated and submerged accidental releases are given in Calculation 12.

Table 14-10 – Calculation 5 - TEDE for other public areas for planned releases from vented fueled experiments at the TS release rate limit for U-235

Location	Fumigation TEDE 24 h (rem)	Eq 8-4 GPM TEDE 96 h (rem)	Eq 8-4 GPM TEDE >96 h (rem)
All	9.7E-04	4.7E-07	1.5E-04
All	1.0E-03	1.5E-05	3.8E-04
All	9.0E-03	3.3E-03	9.0E-03
Withers, Mann	4.1E-04	1.7E-14	3.6E-07
Broughton, Riddick	8.2E-04	1.2E-09	2.0E-05
Patterson, Ricks	9.3E-04	1.3E-07	9.7E-05
DH Hill	9.0E-03	3.3E-03	9.0E-03
Cox	9.1E-04	2.7E-05	4.6E-04
Dabney	2.1E-03	7.8E-04	2.8E-03
Hillsbrgh.St	1.1E-03	7.5E-05	7.5E-04
Talley, Reynolds	8.6E-04	3.8E-05	5.0E-04
Carroll, Syme	6.9E-04	5.9E-05	5.1E-04
North	7.4E-04	2.5E-04	9.8E-04
Maximum, rem	9.0E-03	3.3E-03	9.0E-03
Mass, g	**7.36E-02**	**7.36E-02**	**1.88E-02**

Talley, Reynolds at >96 hours for vented experiment:

$$5.0\times10^{-4}\text{ rem} = 9.0\text{x}10^{-3}\text{ rem }(5.10\times10^{-6}\text{ s m}^{-3} / 9.15\times10^{-5}\text{ s m}^{-3})$$

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. 12-1 for U-235, U-233, Pu-239, and Np-237 using the mass listed in Tables 14-36 and 14-39 from Calculation 12 for an encapsulated and submerged fueled experiment as these are the maximum masses for a fueled experiment.

Because of an accidental release scenario, the maximum mass for an encapsulated fueled experiment is limited by a public exposure time of 24 hours. The sample mass may be irradiated longer. Irradiation times of 1 year (8760 h) and 10 years (87,600 h) were used.

Table 14-11 gives 10 CFR Part 37 Category 2 activity results for Sr-90, Cs-137, and Pm-147 for fueled experiments containing U-235, U-233, Pu-239, and Np-237.

Table 14-11 – Calculation 6 - 10 CFR Part 37 Category 2 Activity Results for Fueled Experiments

Nuclide	Half-life sec	Decay Constant 1/s	Fission Yield Percent Thermal	Fission Yield Percent Non-Thermal	Eq 12-1 Activity Ci	10 CFR Part 37 Category 2 Limit Ci	10 CFR Part 37 Fraction of Limit
1 year U-235; 1.01g							
^{90}Sr	9.07E+08	7.65E-10	5.87	5.60	7.32E-02	3.70E+02	1.98E-04
^{137}Cs	9.47E+08	7.32E-10	3.25	3.76	4.06E-02	2.70E+01	1.50E-03
^{147}Pm	8.26E+07	8.39E-09	2.25	2.14	2.73E-01	1.08E+04	2.53E-05
				TOTAL	3.87E-01		1.73E-03
10 year U-235; 1.01g							
^{90}Sr	9.07E+08	7.65E-10	5.87	5.60	6.59E-01	3.70E+02	1.78E-03
^{137}Cs	9.47E+08	7.32E-10	3.25	3.76	3.67E-01	2.70E+01	1.36E-02
^{147}Pm	8.26E+07	8.39E-09	2.25	2.14	1.09E+00	1.08E+04	1.01E-04
				TOTAL	2.12E+00		1.55E-02
10 year U-233; 0.810 g							
^{90}Sr	9.07E+08	7.65E-10	6.83	6.43	6.18E-01	3.70E+02	1.67E-03
^{137}Cs	9.47E+08	7.32E-10	5.19	5.35	4.64E-01	2.70E+01	1.72E-02
^{147}Pm	8.26E+07	8.39E-09	1.74	1.68	6.87E-01	1.08E+04	6.36E-05
				TOTAL	1.77E+00		1.89E-02
10 year-Pu-239; 0.607 g							
^{90}Sr	9.07E+08	7.65E-10	2.155	2.071	1.86E-01	3.70E+02	5.03E-04
^{137}Cs	9.47E+08	7.32E-10	4.277	4.660	3.66E-01	2.70E+01	1.36E-02
^{147}Pm	8.26E+07	8.39E-09	2.007	1.992	7.58E-01	1.08E+04	7.01E-05
				TOTAL	1.31E+00		1.41E-02
10 year Np237; 1521 g							
^{90}Sr	9.07E+08	7.65E-10	3.525	3.415	3.21E-01	3.70E+02	8.68E-04
^{137}Cs	9.47E+08	7.32E-10	2.322	3.899	3.45E-01	2.70E+01	1.28E-02
^{147}Pm	8.26E+07	8.39E-09	2.502	2.225	9.11E-01	1.08E+04	8.43E-05
				TOTAL	1.58E+00		1.37E-02

Supporting Calculation:

For an irradiation of 1.01 g of U-235 for 8760 h at the neutron fluence rates given in Calculation 1, the Pm-147 activity is:

$$2.73\times10^{-1}\text{ Ci} = [(1.01\text{ g})(6.022\times10^{23}\text{ atoms per mole } / 235\text{ g per mole})] \times 1\text{ decay per atom}$$
$$\times [(585\times10^{-24})(1\times10^{12})(2.25/100)+(571\times10^{-24})(3\times10^{11})(2.14/100)]\ s^{-1}$$
$$\times [1-e^{-(8.39E\text{-}9\text{ per s})(8760\text{ h})(3600\text{ s/h})}]\ [1\text{Ci} / 3.7\times10^{10}\text{ dps}]$$

At higher fluence rates, the mass of U-235 in an encapsulated experiment is reduced to give the same TEDE of 0.01 rem in 24 hours for fumigation conditions outside the reactor building from an accidental release.

As a result, the 10 CFR Part 37 activity and 10 CFR Part 37 fractions are similar for the same irradiation times. Examples are given in Table 14-12. U-233, Pu-239, and Np-237 would also give similar results.

Table 14-12 – Calculation 6 – 10 CFR Part 37 Category 2 Activity Results for Fueled Experiments Using U-235 for Various Irradiation Times and Fluence Rates

Mass Limit g	I-131 Ci	Neutron Fluence $cm^{-2}s^{-1}$		Accident Public TEDE rem	Irradiation Time	Exposure Time Hours	10 CFR Part 37	
		Thermal	Non-thermal				Eq 12-1 Activity, Ci	Fraction of Limit
1.01	1.57	1.00E+12	3.00E+11	0.01	1 y	24	3.87E-01	1.73E-03
1.01	1.57	1.00E+12	3.00E+11	0.01	10 y	24	2.12E+00	1.55E-02
0.0543	1.57	1.60E+13	8.00E+12	0.01	1 y	24	3.82E-01	1.73E-03
0.0543	1.57	1.60E+13	8.00E+12	0.01	10 y	24	2.09E+00	1.55E-02
*133	1.57	1.00E+10	1.00E+08	0.01	1 y	24	4.01E-01	1.73E-03
*133	1.57	1.00E+10	1.00E+08	0.01	10 y	24	2.18E+00	1.55E-02

*NOTE: Maximum U-235 mass requested for fueled experiments is 35 g. 10 CFR Part 37 fraction and activities would be lower by a factor of 35/133.

CALCULATION 7A:
External Dose from the Reactor Building, Overhead Plume, and Reactor Stack for an Encapsulated Fueled Experiment

A fueled experiment accidental release results in external dose due to airborne activity in the reactor building, reactor stack, and an overhead plume. Activity of fission products and halogens from an encapsulated fueled experiment exceeds that from a vented experiment at TS limits.

Table 14-1 was adjusted to give the U-235 source term for an accidental release from an encapsulated experiment. Adjustment was made by multiplying the saturation activities by a ratio of the mass for an encapsulated experiment to the mass for a vented experiment given in Table 14-5; 0.187 g / 0.0736 g.

Results for U-235 reactor building concentrations and the stack and overhead plume activities are given in Table 14-13A.

Tables 14-14A and 14-15A gives the external dose rates from an overhead plume, reactor stack, and reactor building.

Table 14-16A gives the combined external doses (sum of reactor building, overhead plume, and stack) for U-235 at TS Limit for the encapsulated fueled experiment.

Figure 14-2 and Figure 14-3 illustrate the Microshield model used for the overhead plume, reactor stack, and reactor building.

The source term and external dose rates and dose for U-235 were calculated for the accidental release from an encapsulated fueled experiment using U-235 at the TS limit on I-131 release activity of 0.29 Ci. Results are discussed below:

- External dose from the reactor building, overhead plume, and reactor stack were calculated to give 9.1×10^{-4} rem or less to publicly occupied areas outside the reactor building (from Table 14-16A using the 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 5.1×10^{-4} rem based on a minimal distance of 1 m to 10 m in areas occupied for 15 minutes (from Table 14-16A initial rem/h at 1 m; 2.02 mrem/h x 0.25 h = 0.51 mrem).
- 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 27 mrem/h (from Table 14-14A 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 7 mrem or less to roof top occupants.

Table 14-13A – Calculation 7 - U-235 Source Term at TS Limit for Encapsulated Fueled Experiment

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	2.22E-05	1.11E-05	1.16E-06	1.11E-05	4.36E-05	6.29E-04	1.16E-06	4.56E-06	6.58E-05
^{85m}Kr	5.33E-05	3.15E-05	3.83E-06	3.15E-05	1.24E-04	1.79E-03	3.83E-06	1.50E-05	2.17E-04
^{85}Kr	1.17E-05	7.86E-06	1.14E-06	7.86E-06	3.09E-05	4.45E-04	1.14E-06	4.49E-06	6.48E-05
^{87}Kr	1.04E-04	4.60E-05	4.48E-06	4.60E-05	1.81E-04	2.61E-03	4.48E-06	1.76E-05	2.54E-04
^{88}Kr	1.44E-04	7.93E-05	8.96E-06	7.93E-05	3.11E-04	4.49E-03	8.96E-06	3.52E-05	5.07E-04
^{89}Kr	1.79E-04	6.57E-06	5.48E-07	6.57E-06	2.58E-05	3.72E-04	5.48E-07	2.15E-06	3.10E-05
^{131m}Xe	1.61E-06	1.08E-06	1.57E-07	1.08E-06	4.24E-06	6.11E-05	1.57E-07	6.15E-07	8.87E-06
^{133m}Xe	7.79E-06	5.19E-06	7.41E-07	5.19E-06	2.04E-05	2.94E-04	7.41E-07	2.91E-06	4.20E-05
^{133}Xe	2.74E-04	1.83E-04	2.65E-05	1.83E-04	7.20E-04	1.04E-02	2.65E-05	1.04E-04	1.50E-03
^{135m}Xe	4.64E-05	7.37E-06	6.15E-07	7.37E-06	2.89E-05	4.17E-04	6.15E-07	2.41E-06	3.48E-05
^{135}Xe	2.67E-04	1.69E-04	2.22E-05	1.69E-04	6.63E-04	9.56E-03	2.22E-05	8.73E-05	1.26E-03
^{137}Xe	2.49E-04	1.10E-05	9.16E-07	1.10E-05	4.32E-05	6.23E-04	9.16E-07	3.60E-06	5.19E-05
^{138}Xe	2.54E-04	3.76E-05	3.14E-06	3.76E-05	1.48E-04	2.13E-03	3.14E-06	1.23E-05	1.78E-04
Fission Gas Total	1.61E-03								
^{131}I	1.21E-04	8.13E-05	1.18E-05	8.13E-06	3.19E-05	4.60E-04	1.18E-06	4.62E-06	6.66E-05
^{132}I	1.79E-04	9.44E-05	1.03E-05	9.44E-06	3.71E-05	4.93E-06	1.03E-06	4.03E-06	5.82E-05
^{133}I	2.73E-04	1.79E-04	2.49E-05	1.79E-05	7.03E-05	1.01E-03	2.49E-06	9.76E-06	1.41E-04
^{134}I	3.18E-04	1.19E-04	1.09E-05	1.19E-05	4.68E-05	6.75E-04	1.09E-06	4.27E-06	6.17E-05
^{135}I	2.56E-04	1.58E-04	2.01E-05	1.58E-05	6.20E-05	8.95E-04	2.01E-06	7.90E-06	1.14E-04
^{83}Br	2.24E-05	1.19E-05	1.31E-06	1.19E-06	4.67E-06	6.74E-05	1.31E-07	5.12E-07	7.39E-06
^{84}Br	3.99E-05	1.11E-05	9.58E-07	1.11E-06	4.37E-06	6.31E-05	9.58E-08	3.76E-07	5.43E-06
Halogen Total	1.20E-03								

Supporting Calculations

C(0) = [A(∞)(1-r)/V] x Encapsulated Experiment Mass / Vented Experiment Mass
where, A(∞) is taken from Eq. 2-1 as shown in Table 14-1 for U-235
U-235 masses are taken from Table 14-5

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

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

For Kr-87: $C(0) = A(\infty)(1)/V = 2.96\times10^{4}$ µCi x (0.187g/ 0.0221g) / 2.4×10^{9} ml = 1.04×10^{-4} µCi/ml

where $k = 1.52\times10^{-4}$ per s + 1.18×10^{-4} per s = 2.7×10^{-4} per s

2h average <C> = $(1.04\times10^{-4}$ µCi/ml)[(1-exp(-2.7×10^{-4}/s*7200s))/(2.7×10^{-4}/s*7200s)]=4.60×10^{-5} µCi/ml

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

Stack Ci = 3.93 <C> at 24 h for Kr-87 = (3.93)(4.48×10^{-6}) Ci = 1.756×10^{-5} Ci
Overhead Line Ci in calm winds = 56.6 <C> at 2 h for Kr-87 = (56.6)(4.6×10^{-5}) Ci = 2.61×10^{-3} Ci

Table 14-14A – Calculation 7 - External dose rates using Microshield and calculated dose from the reactor building for U-235 at TS Limit for Encapsulated Fueled Experiment

Distance m	Initial rem/h	2 h TEDE rem/h	24 h TEDE rem/h	2 h TEDE rem	24 h TEDE rem
1	2.02E-03	7.72E-04	8.25E-05	1.54E-03	1.98E-03
10	8.78E-04	3.45E-04	3.64E-05	6.90E-04	8.74E-04
20	3.69E-04	1.44E-04	1.53E-05	2.88E-04	3.67E-04
30	1.91E-04	7.24E-05	7.94E-06	1.45E-04	1.91E-04
40	1.14E-04	4.29E-05	4.71E-06	8.58E-05	1.13E-04
50	7.38E-05	2.78E-05	3.05E-06	5.56E-05	7.32E-05
Roof at 1 foot	2.7E-02				

Notes:

- 24 h dose at 30 m $=1.91\times10^{-4}$ rem = (24 h) (7.94×10^{-6} rem/h)
- 1 m to 10 m distance is associated with offices in Burlington labs.
  10 m to 30 m distance is associated with distances to the site boundary.
  Nearby buildings are located at 30 m to 50 m.

Table 14-15A – Calculation 7 - External dose rates and dose calculated using Microshield from overhead plume and reactor stack for U-235 at TS Limit for Encapsulated Fueled Experiment

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	6.03E-07	7.27E-06	1.57E-05	6.53E-08	7.90E-07	2.05E-05
40,0,0	3.83E-07	7.60E-06	1.60E-05	4.13E-08	8.23E-07	2.08E-05
50,0,0	2.59E-07	7.70E-06	1.59E-05	2.80E-08	8.33E-07	2.07E-05
10,0,12	5.07E-06	1.18E-05	3.37E-05	5.50E-07	1.28E-06	4.40E-05
20,0,12	1.66E-06	1.39E-05	3.10E-05	1.79E-07	1.50E-06	4.03E-05
30,0,12	7.93E-07	1.49E-05	3.13E-05	8.63E-08	1.61E-06	4.08E-05
40,0,12	4.57E-07	1.53E-05	3.16E-05	5.00E-08	1.67E-06	4.13E-05
50,0,12	2.93E-07	1.56E-05	3.17E-05	3.17E-08	1.68E-06	4.11E-05

Table 14-16A – Calculation 7 - Combined external doses (sum of reactor building, overhead plume, and stack) for U-235 at TS Limit for Encapsulated Fueled Experiment

Location x,y,z (m)	External Dose in 2 h TEDE (rem)	External Dose in 24 h TEDE (rem)
10,0,12	7.0E-04	9.1E-04
20,0,12	3.1E-04	4.0E-04
30,0,12	1.8E-04	2.3E-04
40,0,12	1.2E-04	1.5E-04
50,0,12	8.7E-05	1.1E-04

Supporting calculation:

2 h TEDE at 50 m (x) downwind from the stack, on the plume centerline (y = 0 m), at a height (z) of 12 m (rooftop) is calculated using results in Tables 14-14A and 14-15A. Results are shown in Table 14-16A.

Example for location 50,0,12 for a 2 h dose: 8.7×10^{-5} rem = ($5.56\text{x}10^{-5} + 3.17\times10^{-5}$) rem

CALCULATION 7B:
External Dose from the Reactor Building, Overhead Plume, and Reactor Stack for an Encapsulated and Submerged Fueled Experiment

For an encapsulated and submerged fueled experiment, the source term is changed from that for an encapsulated fueled experiment. Relatively more fission gases and less halogens are released because of the retention of halogens in the reactor pool.

Calculation 7A was repeated for the source term for an encapsulated and submerged accident release. Results are given in Tables 14-13B through 14-16B:

Table 14-13B – Calculation 7 - U-235 Source Term at TS Limit for Encapsulated and Submerged Fueled Experiment

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	1.20E-04	6.00E-05	6.28E-06	6.00E-05	2.36E-04	3.40E-03	6.28E-06	2.46E-05	3.55E-04
^{85m}Kr	2.88E-04	1.70E-04	2.07E-05	1.70E-04	6.69E-04	9.65E-03	2.07E-05	8.12E-05	1.17E-03
^{85}Kr	6.30E-05	4.25E-05	6.18E-06	4.25E-05	1.67E-04	2.41E-03	6.18E-06	2.43E-05	3.50E-04
^{87}Kr	5.63E-04	2.49E-04	2.42E-05	2.49E-04	9.75E-04	1.41E-02	2.42E-05	9.49E-05	1.37E-03
^{88}Kr	7.77E-04	4.28E-04	4.84E-05	4.28E-04	1.68E-03	2.43E-02	4.84E-05	1.90E-04	2.74E-03
^{89}Kr	9.67E-04	3.55E-05	2.96E-06	3.55E-05	1.39E-04	2.01E-03	2.96E-06	1.16E-05	1.68E-04
^{131m}Xe	8.67E-06	5.83E-06	8.46E-07	5.83E-06	2.29E-05	3.30E-04	8.46E-07	3.32E-06	4.79E-05
^{133m}Xe	4.21E-05	2.80E-05	4.00E-06	2.80E-05	1.10E-04	1.59E-03	4.00E-06	1.57E-05	2.27E-04
^{133}Xe	1.48E-03	9.91E-04	1.43E-04	9.91E-04	3.89E-03	5.61E-02	1.43E-04	5.62E-04	8.10E-03
^{135m}Xe	2.51E-04	3.98E-05	3.32E-06	3.98E-05	1.56E-04	2.25E-03	3.32E-06	1.30E-05	1.88E-04
^{135}Xe	1.44E-03	9.12E-04	1.20E-04	9.12E-04	3.58E-03	5.16E-02	1.20E-04	4.71E-04	6.80E-03
^{137}Xe	1.34E-03	5.94E-05	4.95E-06	5.94E-05	2.33E-04	3.36E-03	4.95E-06	1.94E-05	2.80E-04
^{138}Xe	1.37E-03	2.03E-04	1.70E-05	2.03E-04	7.98E-04	1.15E-02	1.70E-05	6.66E-05	9.61E-04
Fission Gas Total	8.72E-03								
^{131}I	1.96E-05	1.32E-05	1.91E-06	1.32E-06	5.17E-06	7.46E-05	1.91E-07	7.48E-07	1.08E-05
^{132}I	2.90E-05	1.53E-05	1.66E-06	1.53E-06	6.01E-06	4.93E-06	1.66E-07	6.53E-07	9.42E-06
^{133}I	4.43E-05	2.90E-05	4.03E-06	2.90E-06	1.14E-05	1.64E-04	4.03E-07	1.58E-06	2.28E-05
^{134}I	5.15E-05	1.93E-05	1.76E-06	1.93E-06	7.58E-06	1.09E-04	1.76E-07	6.93E-07	9.99E-06
^{135}I	4.15E-05	2.56E-05	3.26E-06	2.56E-06	1.00E-05	1.45E-04	3.26E-07	1.28E-06	1.85E-05
^{83}Br	3.62E-06	1.93E-06	2.12E-07	1.93E-07	7.57E-07	1.09E-05	2.12E-08	8.30E-08	1.20E-06
^{84}Br	6.46E-06	1.81E-06	1.55E-07	1.81E-07	7.09E-07	1.02E-05	1.55E-08	6.10E-08	8.80E-07
Halogen Total	1.96E-04								

Table 14-14B – Calculation 7 - External dose rates using Microshield and calculated dose from the reactor building for U-235 at TS Limit for Encapsulated and Submerged Fueled Experiment

Distance m	Initial rem/h	2 h TEDE rem/h	24 h TEDE rem/h	2 h TEDE rem	24 h TEDE rem
1	5.47E-03	1.38E-03	1.44E-04	2.76E-03	3.46E-03
10	2.24E-03	5.78E-04	6.07E-05	1.16E-03	1.46E-03
20	9.32E-04	2.40E-04	2.51E-05	4.80E-04	6.02E-04
30	4.85E-04	1.25E-04	1.31E-05	2.50E-04	3.14E-04
40	2.89E-04	7.44E-05	7.79E-06	1.49E-04	1.87E-04
50	1.88E-04	4.86E-05	5.08E-06	9.72E-05	1.22E-04
Roof at 1 foot	5.6E-02				

Notes:

1 m to 10 m distance is associated with offices in Burlington labs.
10 m to 30 m distance is associated with distances to the site boundary.
Nearby buildings are located at 30 m to 50 m.

Table 14-15B – Calculation 7 - External dose rates and dose calculated using Microshield from overhead plume and reactor stack for U-235 at TS Limit for Encapsulated and Submerged Fueled Experiment

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	2.66E-06	2.88E-05	6.29E-05	2.97E-07	3.14E-06	8.25E-05
40,0,0	1.61E-06	3.00E-05	6.32E-05	1.77E-07	3.27E-06	8.27E-05
50,0,0	1.06E-06	3.04E-05	6.29E-05	1.17E-07	3.31E-06	8.22E-05
10,0,12	2.03E-05	4.69E-05	1.34E-04	2.22E-06	5.11E-06	1.76E-04
20,0,12	6.59E-06	5.47E-05	1.23E-04	7.23E-07	5.96E-06	1.60E-04
30,0,12	3.14E-06	5.87E-05	1.24E-04	3.44E-07	6.41E-06	1.62E-04
40,0,12	1.80E-06	6.07E-05	1.25E-04	1.97E-07	6.64E-06	1.64E-04
50,0,12	1.15E-06	6.13E-05	1.25E-04	1.26E-07	6.69E-06	1.64E-04

Table 14-16B – Calculation 7 - Combined external doses (sum of reactor building, overhead plume, and stack) for U-235 at TS Limit for Encapsulated and Submerged Fueled Experiment

Location x,y,z (m)	External Dose in 2 h TEDE (rem)	External Dose in 24 h TEDE (rem)
10,0,12	1.3E-03	1.6E-03
20,0,12	6.0E-04	7.6E-04
30,0,12	3.7E-04	4.8E-04
40,0,12	2.7E-04	3.5E-04
50,0,12	2.2E-04	2.9E-04

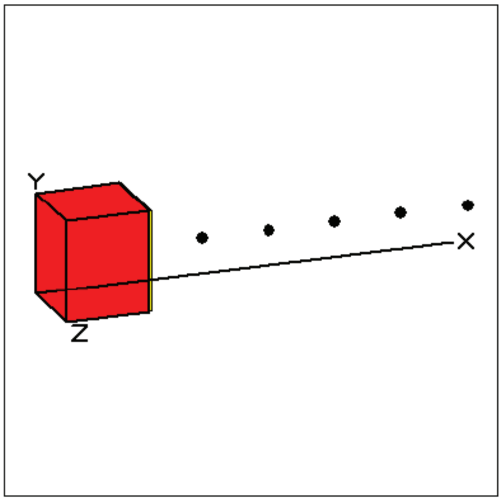
The source term and external dose rates and dose for U-235 were calculated for the accidental release from an encapsulated and submerged fueled experiment using U-235 at the TS limit on I-131 release activity of 1.57 Ci. Results are discussed below:

- External dose from the reactor building, overhead plume, and reactor stack were calculated to give 1.6×10^{-3} rem or less to publicly occupied areas outside the reactor building (from Table 14-16B using the 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 1.4×10^{-3} rem based on a minimal distance of 1 m to 10 m in areas occupied for 15 minutes (from Tables 14-16B initial rem/h at 1 m; 5.47 mrem/h x 0.25 h = 1.4 mrem).
- 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 56 mrem/h (from Table 14-14B 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 14 mrem or less to roof top occupants.

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

Figure 14-2 – Rectangular volume geometry (reactor building with shield is shown)

Geometry: 13 - Rectangular Volume



Source Dimensions

Length	1.3e+3 cm	41 ft 3.0 in
Width	1.3e+3 cm	41 ft 3.0 in
Height	1.5e+3 cm	50 ft

Dose Points

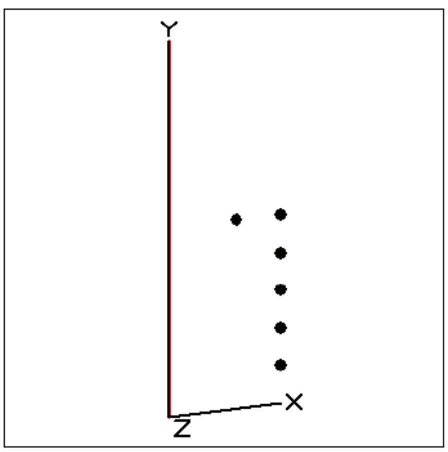
A	X	Y	Z
# 1	1390 cm	762 cm	628.65 cm
	45 ft 7.2 in	25 ft	20 ft 7.5 in
# 2	2290 cm	762 cm	628.65 cm
	75 ft 1.6 in	25 ft	20 ft 7.5 in
# 3	3290 cm	762 cm	628.65 cm
	107 ft 11.3 in	25 ft	20 ft 7.5 in
# 4	4290 cm	762 cm	628.65 cm
	140 ft 9.0 in	25 ft	20 ft 7.5 in
# 5	5290 cm	762 cm	628.65 cm
	173 ft 6.7 in	25 ft	20 ft 7.5 in
# 6	6290 cm	762 cm	628.65 cm
	206 ft 4.4 in	25 ft	20 ft 7.5 in

Shields

Shield Name	Dimension	Material	Density
Source	2409.144 m³	Air	0.00122
Shield 1	.305 m	Concrete	2.35
Air Gap		Air	0.00122

Figure 14-3 – Line source geometry (overhead plume is shown)

Geometry: 2 - Line



Source Dimensions

Length	1.0e+4 cm	328 ft 1.0 in
Angle	90.0°	

Dose Points

A	X	Y	Z
# 1	3000 cm	5000 cm	0 cm
	98 ft 5.1 in	164 ft 0.5 in	0.0 in
# 2	1800 cm	5000 cm	0 cm
	59 ft 0.7 in	164 ft 0.5 in	0.0 in
# 3	3000 cm	4000 cm	0 cm
	98 ft 5.1 in	131 ft 2.8 in	0.0 in
# 4	3000 cm	3000 cm	0 cm
	98 ft 5.1 in	98 ft 5.1 in	0.0 in
# 5	3000 cm	2000 cm	0 cm
	98 ft 5.1 in	65 ft 7.4 in	0.0 in
# 6	3000 cm	1000 cm	0 cm
	98 ft 5.1 in	32 ft 9.7 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Air Gap		Air	0.00122

CALCULATION 8: External dose rates from beamtube exhaust duct for a vented fueled experiment

External dose rates in the beamtube exhaust duct were calculated for a vented fueled experiment with an exhaust rate of 5 lpm (83.3 ml/s) and a decay time of 100 min (decay volume of 500 liters). For the conditions given in Calculations 1 and 2 and at the maximum exposure time of 2240 hours (1 work year). Halogen filter efficiency is 99% for the vented fueled experiment exhaust. At 2240 h or less, the release rate is at a the maximum TS limit.

Table 14-17 and Table 14-18 give the beamtube exhaust duct activity and total dose resulting from U-235 at the TS release rate limit 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 for 2240 h of exposure at 1 m.

Table 14-17 – Calculation 8 - Beamtube Exhaust Duct Activity for U-235 vented fueled Experiment at the TS Release Rate Limit

Nuclide	Saturation Activity (µCi)	Eq 5-8 Decayed, Filtered Vented Release Rate q, (µCi/s)	Eq 8-3 Beam Tube Duct A(duct), (µCi)
83mKr	2.10E+04	1.88E+00	7.52E+00
85mKr	5.03E+04	6.48E+00	2.59E+01
85Kr	1.10E+04	1.84E+00	7.35E+00
87Kr	9.85E+04	6.61E+00	2.64E+01
88Kr	1.36E+05	1.51E+01	6.03E+01
89Kr	1.69E+05	7.83E-09	3.13E-08
131mXe	1.52E+03	2.52E-01	1.01E+00
133mXe	7.36E+03	1.20E+00	4.80E+00
133Xe	2.58E+05	4.27E+01	1.71E+02
135mXe	4.38E+04	7.87E-02	3.15E-01
135Xe	2.52E+05	3.71E+01	1.48E+02
137Xe	2.35E+05	5.16E-07	2.06E-06
138Xe	2.40E+05	2.93E-01	1.17E+00
131I	1.14E+05	1.89E-01	7.57E-01
132I	1.69E+05	1.70E-01	6.82E-01
133I	2.58E+05	4.07E-01	1.63E+00
134I	3.00E+05	1.34E-01	5.36E-01
135I	2.42E+05	3.38E-01	1.35E+00
83Br	2.11E+04	2.17E-02	8.70E-02
84Br	3.76E+04	7.09E-03	2.84E-02
		1.15E+02	4.59E+02

Supporting Calculations:

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

A(duct) = 4 q = 4 [A(∞) / w] p [exp(-λw/p)] where t = w/p

Kr-87: 26.4 µCi = (4s) (9.85×10^{4}µCi / 5.0×10^{5} ml)(83.3 ml/s)[exp($-1.52\times10^{-4} \times 5\times10^{5}/83.3$)], or 2.64×10^{-5} Ci

Using a line source geometry in Microshield with the source term [A(duct)] from Table 14-7 gives the dose rates in Table 14-18:

Table 14-18 – Calculation 8 - Beamtube Exhaust Duct Dose Rates from Microshield and Calculated Dose for U-235 vented fueled experiment at the TS Release Rate Limit

Distance m	TEDE Rate (rem/h)	TEDE (rem)
1	2.18E-05	5.00E-02
2	9.96E-06	2.23E-02
3	6.03E-06	1.37E-02
4	4.10E-06	9.32E-03
5	2.99E-06	6.66E-03
6	2.27E-06	5.00E-03
7	1.78E-06	4.00E-03
8	1.43E-06	3.20E-03
9	1.17E-06	2.63E-03
10	9.79E-07	2.20E-03

Supporting Calculation:

TEDE(rem) = rem/h x 2240 h; e.g. at 3 m 1.37×10^{-2} rem = 6.03×10^{-6} rem/h x 2240 h

CALCULATION 9: Irradiation of 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. Fission product activity is assumed to be at saturation levels.

- Table 14-19 gives the parameter values for experiments containing uranium.
- Table 14-20 gives the time integrated exposures and radiation doses.

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 at a fission rate of 2.0×10^{6} f/s has the following 24 hour doses (see Table 14-21):

- TEDE of 1.0×10^{-3} rem to personnel within the reactor building
- Thyroid TODE of 3.1×10^{-2} rem to personnel within the reactor building
- TEDE of 1.0×10^{-5} rem or less to members of the public outside the reactor building

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

Table 14-19 – 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	1.0E-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			
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	3.15E+07	s	F normal	0.883	m^3/s
			v normal	3.92E-04	1/s
			NG reactor correction	0.1	
Public exposure time	24	hours			

Table 14-19 – 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	2.81E-01
^{85m}Kr	1.61E+04	4.30E-05	1.29E+00	1.36E+00	6.74E-01
^{85}Kr	3.39E+08	2.05E-09	2.83E-01	2.96E-01	1.48E-01
^{87}Kr	4.57E+03	1.52E-04	2.56E+00	2.54E+00	1.32E+00
^{88}Kr	1.02E+04	6.78E-05	3.55E+00	3.43E+00	1.82E+00
^{89}Kr	1.89E+02	3.67E-03	4.51E+00	3.97E+00	2.27E+00
^{131m}Xe	1.03E+06	6.74E-07	4.05E-02	3.54E-02	2.03E-02
^{133m}Xe	1.89E+05	3.66E-06	1.89E-01	1.97E-01	9.86E-02
^{133}Xe	4.53E+05	1.53E-06	6.70E+00	6.71E+00	3.46E+00
^{135m}Xe	9.18E+02	7.55E-04	1.10E+00	1.26E+00	5.87E-01
^{135}Xe	3.28E+04	2.12E-05	6.54E+00	6.58E+00	3.38E+00
^{137}Xe	2.29E+02	3.02E-03	6.13E+00	5.98E+00	3.15E+00
^{138}Xe	8.46E+02	8.19E-04	6.30E+00	6.00E+00	3.22E+00
^{131}I	6.93E+05	1.00E-06	2.89E+00	3.22E+00	1.53E+00
^{132}I	8.26E+03	8.39E-05	4.31E+00	4.66E+00	2.27E+00
^{133}I	7.49E+04	9.26E-06	6.70E+00	6.70E+00	3.46E+00
^{134}I	3.16E+03	2.20E-04	7.83E+00	7.63E+00	4.02E+00
^{135}I	2.37E+04	2.93E-05	6.28E+00	6.27E+00	3.24E+00
^{83}Br	8.64E+03	8.02E-05	5.40E-01	5.76E-01	2.83E-01
^{84}Br	1.91E+03	3.63E-04	9.67E-01	1.01E+00	5.04E-01

Table 14-20 – Calculation 9 - Time Integrated Exposures and Doses from Irradiation of uranium at 2.0×10⁶ 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) Sub DCF	Eq 8-4 Reactor Normal Vent TEDE (rem)	Eq 8-4 Reactor Normal Vent Thyroid TODE (rem)	Eq 8-4 Public Normal Vent TEDE (rem)
^{83m}Kr	3.04E-09	2.29E-11			1.52E-02	4.62E-12	4.62E-12	3.48E-13
^{85m}Kr	7.29E-09	5.50E-11			1.10E+02	8.04E-08	8.04E-08	6.06E-09
^{85}Kr	1.60E-09	1.20E-11			1.74E+00	2.78E-10	2.78E-10	2.09E-11
^{87}Kr	1.43E-08	1.08E-10			5.25E+02	7.50E-07	7.50E-07	5.65E-08
^{88}Kr	1.97E-08	1.48E-10			1.33E+03	2.62E-06	2.62E-06	1.98E-07
^{89}Kr	2.45E-08	1.85E-10			1.20E+03	2.94E-06	2.94E-06	2.22E-07
^{131m}Xe	2.20E-10	1.66E-12			5.48E+00	1.20E-10	1.20E-10	9.07E-12
^{133m}Xe	1.07E-09	8.04E-12			1.99E+01	2.12E-09	2.12E-09	1.60E-10
^{133}Xe	3.74E-08	2.82E-10			2.25E+01	8.41E-08	8.41E-08	6.34E-09
^{135m}Xe	6.35E-09	4.79E-11			2.79E+02	1.77E-07	1.77E-07	1.33E-08
^{135}Xe	3.66E-08	2.76E-10			1.73E+02	6.34E-07	6.34E-07	4.78E-08
^{137}Xe	3.41E-08	2.57E-10			1.10E+02	3.75E-07	3.75E-07	2.82E-08
^{138}Xe	3.48E-08	2.62E-10			7.10E+02	2.47E-06	2.47E-06	1.86E-07
^{131}I	1.66E-08	1.25E-10	3.95E+04	1.30E+06	2.42E+02	6.54E-04	2.15E-02	4.96E-06
^{132}I	2.45E-08	1.85E-10	4.57E+02	7.73E+03	1.49E+03	1.49E-05	1.93E-04	3.60E-07
^{133}I	3.74E-08	2.82E-10	7.02E+03	2.16E+05	3.92E+02	2.64E-04	8.08E-03	2.09E-06
^{134}I	4.35E-08	3.28E-10	1.58E+02	1.28E+03	1.73E+03	1.44E-05	6.31E-05	6.19E-07
^{135}I	3.51E-08	2.64E-10	1.47E+03	3.76E+04	1.06E+03	5.54E-05	1.32E-03	6.71E-07
^{83}Br	3.06E-09	2.31E-11	1.03E+02		5.09E+00	3.18E-07	1.56E-09	2.50E-09
^{84}Br	5.46E-09	4.11E-11	1.01E+02		1.25E+03	1.23E-06	6.84E-07	5.57E-08
TOTAL						1.01-03	3.1E-02	9.5E-06

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

For other fluence rate, the mass of U-235 is adjusted to give a fission rate of 2x10⁶ f/s and TEDE doses less than 0.001 rem inside and outside the reactor building. Examples are given below:

Table 14-21: TEDE from U-235 at 2x10⁶ f/s

Mass g	Thermal Flux	Non-thermal Flux	Total fission rate (f/s)	Reactor Bldg TEDE (rem)	Public TEDE (rem)
1.27E-07	1.00E+13	5.00E+11	2.0E+06	1.0E-03	9.7E-06
8.90E-06	1.00E+11	5.00E+10	2.0E+06	1.0E-03	9.8E-06
1.00E-06	1.00e+12	3.00E+11	2.0E+06	1.0E-03	9.5E-06

CALCULATION 10: Radiation and Experiment Monitor Channel Setpoint Calculations

Radiation Monitor Setpoints

Alarm and alert setpoints for the stack gas, stack exhaust (auxiliary), stack particulate, and vented experiment gas exhaust radiation monitors are provided in TS. 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, alarms occur and actions are taken well before a public TEDE of 0.01 rem is reached.

Stack Gas and Stack Exhaust (Auxiliary) Radiation Monitoring Channels

Alarm and alert setpoints are determined for the stack gas and stack exhaust (auxiliary) radiation monitors based on the more limiting of Emergency Action Levels (EAL) given in the facility Emergency Plan for Ar-41 or from accidental releases of fission gases from vented fueled experiments at TS limits.

The alarm setpoint is set below the initial concentrations of fission gases released from a fueled experiment accident. At the alarm setpoint, the confinement system initiates.

The alert setpoint is set at the sum of maximum concentration of ^{41}Ar released during normal operations and fission gases released from vented fueled experiments at the TS limit. At the alert setpoint, control room annunciation occurs.

Stack Particulate Radiation Monitoring Channel

The stack particulate monitor is equipped with a radioiodine cartridge during fueled experiments as required by the revised TS 3.8.

The stack particulate radiation monitoring alarm set point is based on the more limiting of EAL given in the facility Emergency Plan for Co-60 or from accidental releases of halogens from vented fueled experiments at TS limits. At the alarm setpoint, the confinement system initiates.

The alert setpoint is set below the alarm setpoint and is based on halogens released from vented fueled experiments at TS limits and a fraction of the EAL based on Co-60. At the alert setpoint, control room annunciation occurs.

Vented Fueled Experiment Exhaust Gas Radiation Monitor

The vented fueled experiment exhaust gas radiation monitor is used to measure concentration of fission gases in the experiment exhaust prior to entering the reactor building exhaust system. Kr and Xe fission gases are readily released and are monitored at the experiment exhaust thereby providing an immediate assessment of released activity. The alarm setpoint is based on fission gases released at TS limits. The alert setpoint is based on fission gases released below the TS limit.

At the alarm setpoint, the vented fueled experiment exhaust is halted by isolating the exhaust flow. Both the alarm and alert annunciate in the control room and locally at the experiment. Radiation monitor readings are indicated locally and in the control room.

Stack Gas and Stack Exhaust Radiation Monitor Setpoints

Count rates are used for radiation monitor setpoints, which are dependent on the concentration of the radionuclides released and radiation detector efficiency. Setpoint count rates are determined using facility calibration and surveillance procedures for the accident and abnormal releases.

<u>Alarm Setpoint</u>

An alarm setpoint count rate is used that is below both the EAL and an accidental release of fission gases from a vented fueled experiment. The EAL release limit is 50 ECf for 24 hours, which has an associated dose of 15 mrem in publicly occupied areas outside the reactor building. Release at the EAL:

$$6630 < \mathrm{ECf} > \approx \frac{50\ \mathrm{ECf}}{[(0.00854\ \mathrm{s/m^3})(0.883\ \mathrm{m^3/s})]} \qquad \textit{Equation 14-1}$$

Where, <ECf> is the EC fraction for the nuclide released.
1 ECf of Ar-41 = 1.42×10^{-8} µCi/ml, so 6630 ECf of Ar-41 = 9.4×10^{-5} µCi/ml.

Accidental release of fission gases:

The minimum initial concentration, C(0), of fission gases inside the reactor building is from an accidental release for a vented fueled experiment due to the lower target mass as shown in Table 14-5 . At the TS release rate limit, the vented fueled experiment accident release concentrations, C(0)*I*, and EC fractions, ECf(*i*), are used to determine the total EC fraction exhausted initially by the reactor stack , <ECf stack> using Equation 14-2. Results are given in Tables 14-22, 14-23, and 14-24.

$$< \mathrm{ECf\,stack} \geq \Sigma \left[\frac{\mathrm{C}(0)i}{\mathrm{EC}(i)}\right] = \Sigma(\mathrm{\ ECf})i \qquad \textit{Equation 14-2}$$

Where, C(0)*i* is the initial concentration released for the i^{th} fission gas and is taken from Eq 5-1.
EC(*i*) is an effluent concentration limit determined from the DCF given in Table 14-1.
DCF used are described in Section 8 of the analysis for submersion and inhalation.
Exposure to 1 EC for 8760 h = 0.1 rem, or EC(*i*)=[0.1rem/8760h]/[DCF rem/h per µCi/ml]

Table 14-22 – Calculation 10 – Vented Fueled Experiment Accident Concentrations at the TS Limit

Nuclide	Eq 5-1 C(0) [µCi/ml] U-233	Eq 5-1 C(0) [µCi/ml] U-235	Eq 5-1 C(0) [µCi/ml] Pu-239	Eq 5-1 C(0) [µCi/ml] Np-237
83mKr	1.39E-05	8.75E-06	5.34E-06	5.18E-06
85mKr	2.21E-05	2.10E-05	1.01E-05	7.67E-06
85Kr	2.33E-05	4.59E-06	2.24E-06	7.34E-06
87Kr	5.50E-05	4.10E-05	1.78E-05	1.86E-05
88Kr	7.14E-05	5.66E-05	2.26E-05	2.27E-05
89Kr	7.32E-05	7.05E-05	2.58E-05	2.37E-05
131mXe	5.64E-07	6.32E-07	7.54E-07	3.73E-05
133mXe	2.40E-06	3.07E-06	4.16E-06	6.72E-05
133Xe	8.48E-05	1.08E-04	1.24E-04	7.05E-05
135mXe	1.19E-05	1.83E-05	3.36E-05	7.45E-05
135Xe	8.93E-05	1.05E-04	1.35E-04	7.92E-05
137Xe	8.63E-05	9.79E-05	1.05E-04	6.80E-05
138Xe	7.12E-05	1.00E-04	8.97E-05	5.79E-05

Table 14-23 – Calculation 10 - Vented Fueled Experiment Accident Release at the TS Limit

Nuclide	Eq 14-2 EC (μCi/ml)	Eq 14-2 ECf U-233	Eq 14-2 ECf U-235	Eq 14-2 ECf Pu-239	Eq 14-2 ECf Np-237
83mKr	7.53E-04	1.85E-02	1.16E-02	7.10E-03	6.88E-03
85mKr	1.04E-07	2.13E+02	2.03E+02	9.76E+01	7.41E+01
85Kr	6.56E-06	3.55E+00	6.99E-01	3.41E-01	1.12E+00
87Kr	2.17E-08	2.53E+03	1.89E+03	8.19E+02	8.56E+02
88Kr	8.57E-09	8.33E+03	6.60E+03	2.64E+03	2.65E+03
89Kr	9.51E-09	7.69E+03	7.41E+03	2.71E+03	2.49E+03
131mXe	2.08E-06	2.71E-01	3.03E-01	3.62E-01	1.79E+01
133mXe	5.73E-07	4.19E+00	5.35E+00	7.25E+00	1.17E+02
133Xe	5.08E-07	1.67E+02	2.12E+02	2.44E+02	1.39E+02
135mXe	4.10E-08	2.90E+02	4.47E+02	8.20E+02	1.82E+03
135Xe	6.59E-08	1.35E+03	1.59E+03	2.05E+03	1.20E+03
137Xe	1.04E-07	8.32E+02	9.43E+02	1.01E+03	6.55E+02
138Xe	1.61E-08	4.43E+03	6.22E+03	5.58E+03	3.60E+03

Supporting Calculation:

Kr-88 ECf(i) = 5.66×10^{-5} μCi/ml / 8.57×10^{-9} μCi/ml = 6.60×10^{3} EC

Table 14-24 – Calculation 10 - Vented Fueled Experiment Accident Effluent Concentration at the TS Limit

Parameter	Eq 14-2 <ECf stack> U-233	Eq 14-2 <ECf stack> U-235	Eq 14-2 <ECf stack> Pu-239	Eq 14-2 <ECf stack> Np-237
ΣC(0)i =	6.05E-04	6.35E-04	5.76E-04	5.40E-04
<ECf stack>=	25853	25529	15979	13623

The alarm setpoint is set below the initial concentrations of fission gases released from a fueled experiment accident. At the alarm setpoint, the confinement system is initiated. The lowest <ECf stack> and Total C for fission gases is from Np-237 from Table 14-24. Therefore, the alarm setpoint is based on the lower count rate that meets the following:

- 5.4×10^{-4} μCi/ml, or 13,620 ECf, for an accidental release of fission gases from a vented fueled experiment accident
- 9.4×10^{-5} μCi/ml, or 6630 ECf, from a release of Ar-41.

<u>Alert Setpoint</u>

An alert setpoint count rate is based on the release expected from normal operations. The alert setpoint is based on a count rate above the sum of the releases from fission gases released from a vented fueled experiment and ^{41}Ar releases. The alert setpoint is below the alarm setpoint.

Exceeding the alert setpoint activates an annunciator in the Control Room. Setting the alert setpoint in this way ensures that the Reactor Operator is notified of an abnormal release and allows time for a response. The radiation monitor has a response time of a few seconds and obtains a steady reading within a few minutes. This gives ample time for any mitigating actions to be taken. Assessment of the release may also be made in response to the alert to ensure the 10 CFR Part 20 annual dose constraint of 10 mrem is not exceeded.

^{41}Ar Release:

^{41}Ar has an EC of 1.42×10^{-8} μCi/ml converted from Reference 29. Maximum ^{41}Ar concentration is approximately 1.1×10^{-5} μCi/ml, or 760 ECf, at 1 MW from normal reactor operations.

^{41}Ar Normal Operation Release:

$$760 \text{ ECf} = \left[\frac{1.1 \times 10^{-5} \ \frac{\mu Ci}{ml}}{1.42 \ \times 10^{-8} \ \frac{\mu Ci}{ml} \ per\ EC} \right] \qquad \textit{Equation 14-3}$$

Vented Fueled Experiment Release:

The same equations and analysis used for the vented fueled experiment accident was followed for a planned release of fission gases from a vented fueled experiment. Results are given in Tables 14-25, 14-26, and 14-27. C*i* rather than C(0)*I* was used in Eq 14-2 for the planned vented fueled experiment stack concentrations.

The concentration of fission gases in the reactor stack from a vented fueled experiment is estimated using the fission gas release rate at the TS limit and stack flow rate in normal ventilation.

The fission gas release rates are calculated using EQ 5-5. Concentrations are given in Table 14-25:

Table 14-25 – Calculation 10 –Vented Fueled Experiment Stack Concentration at TS Limits

Nuclide	Eq 5-5 C (μCi/ml) U-233	Eq 5-5 C (μCi/ml) U-235	Eq 5-5 C (μCi/ml) Pu-239	Eq 5-5 C (μCi/ml) Np-237
83mKr	3.39E-06	2.13E-06	1.30E-06	1.26E-06
85mKr	7.75E-06	7.34E-06	3.54E-06	2.69E-06
85Kr	1.06E-05	2.08E-06	1.02E-06	3.33E-06
87Kr	1.00E-05	7.49E-06	3.24E-06	3.40E-06
88Kr	2.15E-05	1.71E-05	6.83E-06	6.86E-06
89Kr	9.21E-15	8.87E-15	3.24E-15	2.98E-15
131mXe	2.55E-07	2.85E-07	3.40E-07	1.69E-05
133mXe	1.06E-06	1.36E-06	1.85E-06	2.98E-05
133Xe	3.81E-05	4.83E-05	5.58E-05	3.17E-05
135mXe	5.81E-08	8.92E-08	1.64E-07	3.64E-07
135Xe	3.56E-05	4.20E-05	5.37E-05	3.16E-05
137Xe	5.15E-13	5.84E-13	6.25E-13	4.06E-13
138Xe	2.37E-07	3.32E-07	2.98E-07	1.92E-07

Table 14-26 – Calculation 10 – Vented Fueled Experiment Release at TS Limits

Nuclide	Eq 14-2 EC µCi/ml	Eq 14-2 ECf U-233	Eq 14-2 ECf U-235	Eq 14-2 ECf Pu-239	Eq 14-2 ECf Np-237
83mKr	7.53E-04	4.50E-03	2.83E-03	1.73E-03	1.68E-03
85mKr	1.04E-07	7.48E+01	7.09E+01	3.42E+01	2.60E+01
85Kr	6.56E-06	1.61E+00	3.17E-01	1.55E-01	5.07E-01
87Kr	2.17E-08	4.62E+02	3.45E+02	1.49E+02	1.56E+02
88Kr	8.57E-09	2.51E+03	1.99E+03	7.97E+02	8.01E+02
89Kr	9.51E-09	9.68E-07	9.32E-07	3.41E-07	3.13E-07
131mXe	2.08E-06	1.22E-01	1.37E-01	1.63E-01	8.09E+00
133mXe	5.73E-07	1.85E+00	2.37E+00	3.22E+00	5.19E+01
133Xe	5.08E-07	7.49E+01	9.51E+01	1.10E+02	6.23E+01
135mXe	4.10E-08	1.42E+00	2.18E+00	4.01E+00	8.89E+00
135Xe	6.59E-08	5.41E+02	6.37E+02	8.15E+02	4.80E+02
137Xe	1.04E-07	4.96E-06	5.63E-06	6.03E-06	3.91E-06
138Xe	1.61E-08	1.47E+01	2.07E+01	1.86E+01	1.20E+01

Table 14-27 – Calculation 10 – Vented Fueled Experiment Effluent Concentration at TS Limits

Parameter	Eq 14-2 <ECf stack> U-233	Eq 14-2 <ECf stack> U-235	Eq 14-2 <ECf stack> Pu-239	Eq 14-2 <ECf stack> Np-237
ΣC(stack)*i* =	1.29E-04	1.29E-04	1.28E-04	1.28E-04
<ECf stack>=	3685	3166	1932	1606

Supporting Calculation:

C(stack) = Release rate / stack flow rate = Q / F and results are given in Table 14-24.

The release rate, Q = q. q is taken from Table 14-2 and calculated similarly for other fissionable materials.

$$C = (0.408\ \text{Ci/h})(1\text{h}/3600\text{s})(1\times10^{6}\ \mu\text{Ci/Ci}) / [(0.883\ \text{m}^3/\text{s})(1\times10^{6}\ \text{ml/m}^3)] = 1.3\times10^{-4}\ \mu\text{Ci/ml}$$

The lowest <ECf stack> and Total C for fission gases is for Np-237 from Table 14-27. Therefore, the alert setpoint is the detector count rate that exceeds the sum of the following:

- 1.3×10^{-4} µCi/ml, or 1606 ECf, for a planned release of fission gases from a vented fueled experiment accident plus
- 1.1×10^{-5} µCi/ml, or 760 ECf, from a release of ^{41}Ar.

This gives a minimum total of 2366 ECf for a combination of Ar-41 and a planned release of fission gases from a vented fueled experiment.

At 2370 ECf, the TEDE to a member of the public outside the reactor building under fumigation conditions for 24 hours is 5 mrem.

$$4.9\times10^{-3}\ \text{rem} \sim [2366\ \text{ECf}][24\ \text{h}]\ [0.1\ \text{rem/ECf}/8760\ \text{h}][0.00854\ \text{s/m}^3][0.883\ \text{m}^3/\text{s}]$$

The alert level initiates an annunciator in the Control Room and would prompt the reactor operator to take action. If the release were to occur for 2 hours, the TEDE is less than 0.5 mrem.

If the alert level of 2370 EC were to occur for 4600 h (over 190 days) in a year, the annual TEDE would be 10 mrem using sector averaged weather conditions.

$$1x10^{-2} \text{ rem} = [2366 \text{ ECf}][4600 \text{ h}][9.15x10^{-5} \text{ s/m}^3][0.883 \text{ m}^3\text{/s}]\ [0.1 \text{ rem/ECf/8760h}]$$

Therefore, an alert at 2360 ECf for fission gases and Ar-41 allows early detection of the annual dose constraint for airborne effluent.

Stack Particulate Radiation Monitor Setpoints:

The alarm and alert setpoints for the stack particulate radiation monitor are based on releases of fission halogens from a fueled experiment or ^{60}Co. Accidental releases are associated with the alarm setpoint and abnormal releases are associated with the alert setpoint. Exceeding the alarm setpoint activates the Confinement System. Exceeding the alert setpoint sounds an annunciator in the Control Room. The stack particulate monitor is equipped with both particulate and radioiodine (halogen) air sample media. Activity deposited on the both sample media would be detected by the stack particulate radiation monitor.

Accidental releases are associated with a release at the Emergency Action Level (EAL) as stated in the facility Emergency Plan or a fueled experiment failure at the TS limit. Abnormal releases are from fission halogens from a vented fueled experiment above the TS limit.

Count rates are used for radiation monitor setpoints, which are dependent on the concentration of the radionuclides released, buildup of activity on the sample filter media, and radiation detection efficiency. Setpoint count rates are determined using facility calibration and surveillance procedures for accident and abnormal releases. Setpoints are related to and are determined from the concentration of halogens being released.

<u>Alarm Setpoint</u>

An alarm setpoint count rate is used that is below both the EAL or an accidental release of fission halogens from a fueled experiment and above the release of halogens from a vented fueled experiment at the TS limit.

The alarm setpoint for the stack particulate radiation monitor is based on the more limiting of an EAL of 100 ECf for 24 hours with a public dose of 15 mrem or an accidental release of halogens from a fueled experiment at the TS limit. The TEDE from halogens released less than 3 mrem outside the reactor building.

Using the fumigation [X/Q] value of $8.54x10^{-3}$ s/m^3 for occupied areas and an exposure time of 24 hours and the normal stack exhaust rate of 0.883 m^3/s gives the stack particulate monitor setpoints:

EAL Release for Co-60:

$$13{,}261 \text{ ECf} = \frac{100 \text{ ECf}}{[(0.00854\,\text{s/m}^3)(0.883\,\text{m}^3\text{/s})]} \qquad \textit{Equation 14-4}$$

EC for Co-60 is $5x10^{-11}$ µCi/ml. 6630 ECf of Co-60 = $6.6x10^{-7}$ µCi/ml.

Fueled Experiment Release:

In an accident release for a vented fueled experiment, halogens are released into the reactor building free air volume and exhausted by the ventilation system. The released activity in the reactor ventilation system is filtered after the alarm setpoint is reached and the confinement system is initiated.

In a planned release for a vented fueled experiment, exhausted halogens from the experiment are filtered with a combined retention of 99 percent $[1-(1-0.9)^2 = 0.99]$ and decayed for 100 minutes or more prior to being released into the reactor building exhaust system.

The reactor ventilation system is sampled with activity being accumulated on a cartridge. Typical filter retention for halogens on the sample cartridge at low flow rates is 0.9 or more.

EQ 5-1, EQ 5-5 and EQ 14-2 were performed for halogens released in an accident and a planned release for vented fueled experiments at TS limits. C*i* rather than C(0)*I* was used in Eq 14-2 for the planned vented fueled experiment stack concentrations. EC(*i*) were taken from 10 CFR Part 20 Appendix B rather than the DCF listed in Table 14-1.

Results are given in Table 14-28:

Table 14-28 – Calculation 10 –Stack Halogen Releases for Accident and Planned Vented Fueled Experiments at TS Limits

Nuclide	**EC µCi/ml**	**Eq 5-1 C(0) U-233 µCi/ml**	**Eq 5-1 C(0) U-235 µCi/ml**	**Eq 5-1 C(0) Pu-239 µCi/ml**	**Eq 5-1 C(0) Np-237 µCi/ml**
131I	1.46E-10	5.18E-05	4.76E-05	6.85E-05	3.90E-05
132I	3.07E-09	7.05E-05	7.05E-05	9.53E-05	5.27E-05
133I	7.77E-10	8.48E-05	1.08E-04	1.23E-04	7.04E-05
134I	3.24E-09	8.65E-05	1.25E-04	1.30E-04	7.59E-05
135I	2.33E-09	7.21E-05	1.01E-04	1.14E-04	7.33E-05
83Br	5.26E-08	1.39E-05	8.80E-06	5.34E-06	5.18E-06
84Br	4.21E-09	2.30E-05	1.57E-05	7.76E-06	7.58E-06
Accident	ΣC(0)*i* =	4.03E-04	4.76E-04	5.45E-04	3.24E-04
Release	<ECf stack> =	5.51E+05	5.75E+05	7.52E+05	4.33E+05
		Eq 5-5 C	**Eq 5-5 C**	**Eq 5-5 C**	**Eq 5-5 C**
131I	1.46E-10	2.33E-07	2.15E-07	3.09E-07	1.76E-07
134I	3.24E-09	1.05E-07	1.52E-07	1.58E-07	9.21E-08
135I	2.33E-09	2.74E-07	3.83E-07	4.34E-07	2.79E-07
83Br	5.26E-08	3.89E-08	2.46E-08	1.50E-08	1.45E-08
84Br	4.21E-09	1.18E-08	8.04E-09	3.97E-09	3.88E-09
Planned, Vented	ΣC(0)*i* =	1.22E-06	1.44E-06	1.71E-06	1.01E-06
Release	<ECf stack> =	1587	1610	2161	1236

From Table 14-28, initial accident release concentrations, C(0), are significantly higher than stack concentrations, C, from a planned release for vented fueled experiments. The reactor stack sampling unit uses fixed filters for particulates and halogens. Sampled activity accumulates on sample filters. Higher concentrations result in higher activity accumulation. Filters are analyzed by gamma spectroscopy weekly.

The setpoint is based on a planned release from a vented fueled experiment at the TS limit. Any activity above the planned release is abnormal and may be due to a high release rate, accident, or leakage into the reactor building air space. An accident at TS limits is readily detectable. The stack particulate radiation monitoring alarm set point is based on the more limiting of EAL given in the facility Emergency Plan for Co-60 or from accidental releases of halogens from vented or encapsulated fueled experiments at TS limits. At the alarm setpoint, the confinement system is initiated.

The alarm setpoint is based on the lower count rate from the activity buildup on sample filter media based on the following concentrations:

- 6.6×10^{-7} µCi/ml, or 13,260 ECf based on ^{60}Co
- 1.7×10^{-6} µCi/ml, or 2160 ECf for halogen activity from a planned release for a vented fueled experiment

Alert Setpoint

The alert setpoint is lower than the alarm setpoint and is based on the lower count rate from the activity buildup on sample filter media based on the following concentrations:

- 4.4×10^{-7} µCi/ml, or 8800 ECf based on ^{60}Co
- 1.0×10^{-6} µCi/ml, or 1235 ECf for halogen activity from a planned release for a vented fueled

8800 ECf of Co-60 equates to 4.4×10^{-7} µCi/ml with a public TEDE of 10 mrem from an accident and is below the EAL.

Exceeding the alert setpoint sounds an annunciator in the Control Room making the Reactor Operator aware of a potentially abnormal release. Abnormal releases are promptly analyzed allowing sufficient time to determine if TS limits are being met. The radiation monitor has a response time of a few minutes, which gives sufficient time for any mitigating actions to be taken before TS limits or the 10 CFR Part 20 annual dose constraint of 10 mrem is reached.

Table 14-29: Setpoints for Required Reactor Stack Radiation Monitors

Monitor	Alert Setpoint	Alarm Setpoint
Stack Gas	Sum of count rate from 1.1×10^{-5} µCi/ml of Ar-41 and 1.3×10^{-4} µCi/ml of Fission Gas	Lower count rate from 9.4×10^{-5} µCi/ml of Ar-41 or 5.4×10^{-4} µCi/ml of Fission Gas
Stack Particulate	Lower count rate from 4.4×10^{-7} µCi/ml of Co-60 or 1.0×10^{-6} µCi/ml of Fission Halogens	Lower count rate from 6.6×10^{-7} µCi/ml of Co-60 or 1.7×10^{-6} µCi/ml of Fission Halogens

Vented Fueled Experiment Exhaust Radiation Monitor Setpoints

The purpose of this monitor is to detect and isolate releases above TS limits. The vented fueled experiment exhaust radiation monitor setpoints are based on the TS release rate limit for fission gases. This monitor is located at the experiment exhaust before the connection to the reactor building ventilation exhaust. The alarm setpoint isolates the experiment exhaust and is set above the alert setpoint. The alert setpoint provides Control Room annunciation.

Fission gases are monitored at the exhaust line or in an exhaust gas container; e.g. Marinelli beaker or a delay tank. The VFE monitor setpoint is therefore based on either concentration or activity in the detection volume. Calibration and surveillance procedures are used to determine the count rate for the alarm and alert setpoints; i.e. the count rate per unit concentration or activity is determined during calibration.

The release rate and setpoints depend on the experiment irradiation time. The release rate at TS limits ranges from 0.407 Ci/h (113 µCi/s) to 0.0601 Ci/h (16.7 µCi/s). Concentration of fission gases is given by the release rate divided by the experiment exhaust rate; q/p, from EQ 5-5 and 5-6.

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 100 minutes is required, i.e. the 500 liter decay volume is vented at a flow rate of 5 lpm. The vented fueled experiment gas flow rate monitor setpoint would therefore be set at 5 lpm for this experiment to assure that the minimum decay time of 100 minutes is met.

CALCULATION 11: Calculations for Additional Fissionable Materials

Calculations for other fissionable materials were made as described in this analysis to determine which fissionable materials are restrictive for common mixtures. 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.

Restrictive Fissionable Materials

Results for vented experiments and encapsulated experiments are provided in Tables 14-30 and 14-31 for different fissionable materials. The higher, i.e. more conservative, rem per unit fission rate (f/s) was used to determine restrictive fissionable materials for mixtures.

Conditions and Data

Conditions for vented experiments were at an exhaust flow rate of 5 lpm and decay time of 100 minutes, and fluence rates of 1×10^{12} thermal and 3×10^{11} non-thermal were used for both vented and encapsulated experiments.

Supporting data used in these calculations are given in Table 14-32 through Table 14-35.

Table 14-30 - Comparison of Results for Mixtures of Fissionable Materials

Fissionable Material	Encapsulated Accident rem per f/s	Vented Public rem per f/s	Restrictive Material for Mixture
^{232}U	1.17E-14	3.15E-14	U-233
^{233}U	3.09E-14	7.94E-14	U-233
^{234}U	2.56E-14	5.63E-14	U-235
^{235}U	2.76E-14	6.30E-14	U-235
^{236}U	2.69E-14	5.54E-14	U-235
^{238}U	2.79E-14	4.86E-14	U-235
^{237}Np	2.94E-14	5.36E-14	Np-237
^{238}Np	2.78E-14	4.65E-14	Np-237
^{238}Pu	2.93E-14	4.62E-14	Pu-239
^{239}Pu	3.02E-14	4.60E-14	Pu-239
^{240}Pu	2.84E-14	4.13E-14	Pu-239
^{241}Pu	2.74E-14	4.12E-14	Pu-239
^{242}Pu	2.59E-14	3.73E-14	Pu-239

Restrictive fissionable materials were determined from radiation dose per unit fission rate from Table 14-30 Results for these mixtures are given in Table 14-31 for common mixtures:

Table 14-31 – Restrictive Fissionable Materials

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

Data for Other Fissionable Materials

Table 14-32 – 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 14-33 – Cumulative Fission Yields per 100 Fissions and Fission Cross Sections (barns)

	^{235}U		^{233}U		^{239}Pu		^{237}Np	
Nuclide	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 14-34 – 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 14-35 – 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

CALCULATION 12: Analysis for an Encapsulated and Submerged Fueled Experiment

For an encapsulated and submerged experiment in an irradiation facility in the reactor pool, an accidental release is assumed to occur during the irradiation. All of the fission gases and halogens are then assumed to enter the reactor pool. All of the fission gases from the reactor pool are assumed to enter the reactor building free air volume (reactor bay air space). However, only three percent of the halogens escape the reactor pool to enter the reactor air space. Three percent release of halogens is taken from the NCSU PULSTAR SAR (1997) which states 97 percent of the iodine and bromine isotopes are assumed to be retained in the reactor water as cited from " Robertson, R.F.S., et al, Fuel Defect test - Borax IV, ANL 5862, October 1959".

Radiation doses for an encapsulated and submerged experiment with a release of 3 percent of halogens from the reactor pool giving a public dose of 0.01 rem under fumigation conditions are given in Tables 14-36 and 14-37:

Table 14-36 – Doses from an Encapsulated and Submerged Experiment for U-235

Experiment	I-131 Ci	U-235 Mass, g	Nuclides	Reactor Bay TEDE rem	Reactor Bay Thyroid TODE rem	Public TEDE rem
Encapsulated and submerged	1.57	1.01E+00	All	1.49E-01	3.60E+00	1.00E-02
			Fission Gas	3.44E-02	3.44E-02	8.67E-03
			Halogens	1.15E-01	3.57E+00	1.36E-03
Encapsulated	0.29	1.87E-01	All	7.17E-01	2.20E+01	1.00E-02
			Fission Gas	6.36E-03	6.36E-03	1.60E-03
			Halogens	7.10E-01	2.20E+01	8.39E-03
Vented		7.36E-02	All	2.82E-01	8.67E+00	3.93E-03
			Fission Gas	2.50E-03	2.50E-03	6.32E-04
			Halogens	2.80E-01	8.67E+00	3.30E-03

Comparison is made in Table 14-36 for results from vented and encapsulated experiments that are not submerged in the reactor pool for U-235.

Table 14-37 – Doses from an Encapsulated and Submerged Experiment for Other Fissionable Materials

Experiment	I-131 Ci	Mass, g	Nuclides	Reactor Bay TEDE rem	Reactor Bay Thyroid TODE rem	Public TEDE rem
U-233	1.58	0.815	All	1.37E-01	3.31E+00	1.00E-02
			Fission Gas	3.24E-02	3.24E-02	8.80E-03
			Halogen	1.05E-01	3.28E+00	1.22E-03
Pu-239	3.08	1.19	All	2.40E-01	6.60E+00	1.00E-02
			Fission Gas	2.99E-02	2.99E-02	7.58E-03
			Halogen	2.10E-01	6.57E+00	2.45E-03
Np-237	2.26	2190	All	1.88E-01	4.87E+00	1.00E-02
			Fission Gas	3.29E-02	3.29E-02	8.19E-03
			Halogen	1.55E-01	4.84E+00	1.81E-03

Supporting Calculations:

Results given in Section 14 Calculation 2 Table 14-4 for Pu-239 are modified for a halogen release of 3 percent and a public TEDE of 0.01 rem. Results were also determined for U-233, U-235, and Np-237. Results are then summarized in Table 14-36 and 14-37 for encapsulated and submerged fueled experiments.

Table 14-38 – Doses from Ventilation Modes for an Encapsulated and Submerged Experiment for Pu-239

	Reactor - Confine TEDE rem	Reactor - Confine Thyroid TODE rem	Public- Confine TEDE rem	Reactor - Normal TEDE rem	Reactor – Normal Thyroid TODE rem	Public - Normal TEDE rem
Table 14-4 results for 100% Halogen Release and I-131 = 0.29 Ci:						
Total =	2.26E-01	7.05E+00	4.57E-03	4.36E-01	1.36E+01	3.83E-03
Fission Gas =	9.98E-04	9.98E-04	5.76E-04	1.82E-03	1.82E-03	1.37E-04
Halogen =	2.25E-01	7.04E+00	3.99E-03	4.34E-01	1.36E+01	3.70E-03
Modified Table 14-4 results for 3% Halogen Release at I-131 = 0.29 Ci:						
Total =	7.76E-03	2.12E-01	6.96E-04	1.48E-02	4.09E-01	2.48E-04
Fission Gas =	9.98E-04	9.98E-04	5.76E-04	1.82E-03	1.82E-03	1.37E-04
Halogen =	6.76E-03	2.11E-01	1.20E-04	1.30E-02	4.07E-01	1.11E-04
Modified Table 14-4 results for 3% Halogen Release and a Public TEDE = 0.01 rem:						
Total =	8.24E-02	2.26E+00	7.39E-03	1.58E-01	4.35E+00	2.64E-03
Fission Gas =	1.06E-02	1.06E-02	6.12E-03	1.93E-02	1.93E-02	1.46E-03
Halogen =	7.18E-02	2.25E+00	1.27E-03	1.38E-01	4.33E+00	1.18E-03

Reactor Halogen dose at 3 percent release from the reactor pool with I-131 = 0.29 Ci in normal ventilation mode:

1.30E-02 rem = 4.34E-01 rem x 0.03

Public TEDE rem from halogens at 3 percent release from the reactor pool and a public TEDE of 0.01 rem:

2.45E-03 rem = 1.27E-03 rem + 1.18E-03 rem

For a public TEDE of 0.01 rem, the mass for an encapsulated experiment is increased by the ratio of 10.6 [10 mrem/(0.699 mrem + 0.248 mrem)] to give 1.19 g (10.6 x 0.112 g).

Saturation I-131 activity increases by the same ratio to give 3.08 Ci (0.29 Ci x 10.6) for encapsulated and submerged experiments.

Table 14-39 – Doses from an Encapsulated and Submerged Experiment at TS Limit for I-131 Activity

Experiment	I-131 Ci	Mass, g	Nuclides	Reactor Bay TEDE rem	Reactor Bay Thyroid TODE rem	Public TEDE rem
U-233	1.57	0.810	All	1.36E-01	3.29E+00	9.94E-03
			Fission Gas	3.22E-02	3.22E-02	8.74E-03
			Halogen	1.04E-01	3.26E+00	1.21E-03
Pu-239	1.57	0.607	All	1.22E-01	3.36E+00	5.10E-03
			Fission Gas	1.52E-02	1.52E-02	3.86E-03
			Halogen	1.07E-01	3.35E+00	1.25E-03
Np-237	1.57	1521	All	1.31E-01	3.38E+00	6.95E-03
			Fission Gas	2.29E-02	2.29E-02	5.69E-03
			Halogen	1.08E-01	3.36E+00	1.26E-03

Public TEDE for Pu-239 = 5.1E-03 rem = (1.57 Ci / 3.08 Ci)(0.01 rem)

Discussion and Conclusion:

From the above results, U-235 has the lowest I-131 saturation activity of 1.57 Ci. I-131 activity for U-233 is similar to U-235. I-131 activity at 1.57 Ci is lower than those for Pu-239 and Np-237.

Results for submerged and encapsulated fueled experiments as calculated do not exceed a public TEDE of 0.01 rem, a reactor bay TEDE of 0.188 rem (< 1 rem) and a reactor bay thyroid TODE of 6.6 rem (< 25 rem). At a lower I-131 activity, the radiation doses would be lower.

Therefore, using I-131 activity of 1.57 Ci as a limit for encapsulated and submerged fueled experiments for the fissionable materials requested (U, Pu, and Np) will ensure the public TEDE does not exceed 0.01 rem and the reactor bay TEDE is less than 1 rem and reactor bay thyroid TODE is less than 25 rem.

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.
- Worse 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 worse 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:

- Released activity and radiation dose is prospectively analyzed and experimental conditions are set to meet TS requirements. It is important to note that compliance with the TS limits 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 0.01 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 released activity at the concentrations that meet TS limits. 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 TS limit or a TEDE of 0.01 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) worse 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 0.01 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 an additional factor of 2 may also apply to dose assessment if credit is not 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 were calculated for vented and encapsulated fueled experiments.

TS limit for planned releases from vented fueled experiments is based on a radiation dose of ten percent of the annual radiation dose limit for members of the public outside the reactor building; i.e. a TEDE of 0.01 rem. TEDE from accidental releases for vented fueled experiments was less than 0.01 rem.

TS limit for vented fueled experiments is based on the release of fission gases over the irradiation period. For irradiation times up to 93 days (2240 h), a fixed release rate is used for vented fueled experiments due to a 24 hour TEDE of 0.01 rem under worse case weather conditions (fumigation).

For longer irradiation times, the released activity of fission gases is divided by the irradiation time to give a release rate for fueled experiments. The total release activity for vented fueled experiments longer than 93 days (2240 hours) is limited to 975 Ci.

TS limits for encapsulated fueled experiments is based on an accidental release due to a 24 hour TEDE of 0.01 rem under worse case weather conditions (fumigation). Separate TS limits for I-131 saturation activity for encapsulated and encapsulated and submerged fueled experiments are used. I-131 is indicative of all fission gases and all halogens released from an encapsulation failure.

Other TS conditions are given to meet the pubic TEDE of 0.01 rem from vented fueled experiments. These include filtration and decay of vented fueled experiment exhaust, monitoring of the experiment exhaust for radioactivity and flow rate, and setpoint changes to reactor stack airborne effluent monitors.

Public dose from vented fueled experiments at TS release rate limits and encapsulated fueled experiments were calculated for restrictive fissionable materials to be well below regulatory limits. Compliance with TS limits and public dose limits is provided by measured data from the vented fueled experiment exhaust and reactor stack.

TS require an experiment specific analysis to be performed and reviewed prior to performing a fueled experiment. The experiment specific analysis is used to prospectively determine experimental irradiation conditions that meet TS limits. TS limits were based on conservative results for various fissionable materials. For mixtures of fissionable materials this approach results in public dose being at or below a TEDE of 0.01 rem for vented fueled experiments and a TEDE of 0.01 rem for encapsulated fueled experiments.

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. Automatic isolation of the fueled experiment occurs if the exhaust radiation or flow monitor readings exceed setpoints. 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 activity of Sr-90, Cs-137, and Pm-147. At the TS limits for the fluence rates achievable in the experimental facilities, the activity produced are less than two percent of 10 CFR Part 37 limits.

In all experimental cases analyzed, the TEDE in publicly occupied areas outside the reactor building does not exceed 0.01 rem for vented fueled experiments and 0.01 rem for encapsulated fueled experiments. The radiation dose constraint given in 10 CFR Part 20 is at a TEDE of 0.01 rem and the annual radiation dose limit for members of the public given in 10 CFR Part 20 is at a TEDE of 0.1 rem. **Given the low**

radiation dose and the conservatisms applied as detailed in this analysis, radiation dose to the public from fueled experiments performed at the facility is well below limits and indiscernible from background radiation dose.

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ATTACHMENT 2:
Technical Specification Changes for Fueled Experiments LAR

1.2.7 **Control Rod**: A control rod is a neutron absorbing blade having an in-line drive which is magnetically coupled and has SCRAM capability.

1.2.8 **Excess Reactivity**: Excess reactivity is that amount of reactivity that would exist if all control rods (and Shim Rod) were fully withdrawn from the point where the reactor is exactly critical ($k_{eff}=1$).

1.2.9 **Experiment**: Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beam tube or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be a part of their design. Specific categories of experiments include:

a. **Tried Experiment**: Tried experiments are those experiments that have been previously performed in this reactor. Specifically, a tried experiment has similar size, shape, composition and location of an experiment previously approved and performed in the reactor.

b. **Secured Experiment**: A secured experiment is any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

c. **Non-Secured Experiment**: A non-secured experiment is an experiment that does not meet the criteria for being a "secured" experiment.

d. **Movable Experiment**: A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

e. **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.~~contains fissionable materi~~

1.2.10 Experimental Facilities: Experimental facilities are facilities used to perform experiments. They include beam tubes, thermal columns, void tanks, pneumatic transfer systems, in-core facilities at single-assembly positions, out-of-core irradiation facilities, and the bulk irradiation facility.

1.2.11 Limiting Condition for Operation: Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility (10 CFR 50.36).

1.2.12 Limiting Safety System Setting: Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded (10 CFR 50.36).

1.2.13 Measured Value: The measured value is the value of a parameter as it appears on the output of a channel.

1.2.14 Operable: Operable means a component or system is capable of performing its intended function.

1.2.15 Operating: Operating means a component or system is performing its intended function.

1.2.16 pcm: A unit of reactivity that is the abbreviation for "percent millirho" and is equal to 10^{-5} $\Delta k/k$ reactivity. For example, 1000 pcm is equal to 1.0% $\Delta k/k$.

3.5 Radiation Monitoring Equipment

Applicability

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective

To assure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specification

The reactor shall not be operated nor shall irradiated fuel or irradiated fueled experiments be moved within the reactor building unless the radiation monitoring equipment listed below and in Table 3.5-1 ~~is~~ are operable.[(1)(2)(3)(7)]

a. Three fixed area monitors operating in the Reactor Building with their setpoints as listed in Table 3.5-1.[(1)(3)(4)]

b. ~~Particulate and gas~~Stack Gas and Stack Particulate building exhaust monitors continuously sampling air in the facility exhaust stack with their setpoints as listed in Table 3.5-1.[(1)(3)(4)]

c. The Radiation Rack Recorder.[(5)]

Table 3.5-1: Required Radiation Area Monitors		
Monitor	Alert Setpoint	Alarm Setpoint
Control Room	≤ 2 mR/hr	≤ 5 mR/hr
Over-the-Pool	≤ 5 mR/hr	≤ 100 mR/hr
West Wall	≤ 5 mR/hr	≤ 100 mR/hr
Stack Gas	Sum of count rate from 1.1×10^{-5} µCi/ml of Ar-41 and 1.3×10^{-4} µCi/ml of Fission Gas [(6)(9)] ~~□ 1000 Ar-41 AEC[(6)]~~	Lower count rate from 9.4×10^{-5} µCi/ml of Ar-41 or 5.4×10^{-4} µCi/ml of Fission Gas [(6)(9)] ~~□ 5,000 Ar-41AEC[(6)]~~
Stack Particulate	Lower count rate from 4.4×10^{-7} µCi/ml of Co-60 or 1.0×10^{-6} µCi/ml of Fission Halogens [(6)(9)] ~~□ 1000 Co-60 AEC[(6)]~~	Lower count rate from 6.6×10^{-7} µCi/ml of Co-60 or 1.7×10^{-6} µCi/ml of Fission Halogens [(6)(9)] ~~□ 5,000 Co-60 AEC[(6)]~~

d. Vented fueled experiment exhaust gas radiation monitor continuously monitoring the experiment exhaust gas.(7)(8)

e. Vented fueled experiment flow rate monitor continuously monitoring the experiment exhaust gas flow.(7)(8)

(1) For periods of time, not to exceed ninety days, for maintenance to the radiation monitoring channel, the intent of this specification will be satisfied if one of the installed channels is replaced with a gamma-sensitive instrument which has its own alarm audible or observable in the control room. Refer to SAR Section 5.

(2) The Over-the-Pool Monitor may be bypassed for less than two minutes during return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

(3) Stack Gas and Particulate are based on the AEC quantities present in the ventilation flow stream as it exits the stack. Refer to SAR Section 10 for setpoint bases for the radiation monitoring equipment.

(4) May be bypassed for less than one minute immediately after starting the pneumatic blower system.

(5) During repair and/or maintenance of the recorder not to exceed 90 days, the specified area and effluent monitor readings shall be recorded manually at a nominal interval of 30 minutes when the reactor is not shutdown. Refer to SAR Section 5.

(6) Co-60 and Ar-41 are radionuclides of concern without a fueled experiment in progress.

(7) Monitors for vented fueled experiments are only required to be operable while the experiment is in operation.

(8) Vented fueled experiment exhaust gas radiation and flow monitor setpoints meet Specification 3.8.d.iv and isolate the experiment exhaust when exceeded.

(9) Fission gases and halogens are released from vented fueled experiments or fueled experiment accidents.

~~[6] Airborne Effluent Concentrations (AEC) values from 10 CFR Part 20 Appendix B, Table 2~~

Bases

A continued evaluation of the radiation levels within the Reactor Building will be made to assure the safety of personnel. This is accomplished by the area monitoring system of the type described in Section 5 of the Safety Analysis Report (SAR).

Evaluation of the continued discharge air to the environment will be made using the information recorded from the stack particulate and stack gas monitors.

When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, the building will be automatically placed in confinement as described in SAR Section 5.

To prevent unnecessary initiation of the evacuation confinement system during the return of a pneumatic capsule from the core to the unloading station or during removal of experiments from the reactor pool, the Over-the-Pool Monitor may be bypassed during the specified time interval. Refer to SAR Section 5.

Stack gas and stack particulate setpoints are based on the Notification of Unusual Event Emergency Action Level (EAL) for Ar-41 and Co-60, respectively, during normal operation with no vented fueled experiments being performed.

Radiation dose for EAL are higher than those for fueled experiments. While fueled experiments are performed, the stack gas and stack particulate monitor setpoints are changed to meet setpoints listed in Table 3.5-1.

In addition, the exhaust fission gases and exhaust flow rate from vented fueled experiments are monitored in accordance with Specification 3.8. Upon reaching an alarm setpoint from the vented fueled experiment exhaust gas radiation or flow monitors, the vented fueled experiment exhaust is automatically isolated. Radiation monitor setpoints are analyzed as described in the documentation presented in the Fueled Experiment Analysis Report for TS Amendment 19.

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. The mass, fission rate and power are limited as indicated in Figure 3.8-1 and Table 3.8-1.~~

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

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.iv.1.

d. Fissionable material(s) in a fueled experiment shall meet the following:

   i. Physical form shall be solid, powder, or liquid.

   ii. Limited to any mixture of U, Pu, and Np.

   iii. Encapsulated fueled experiments are limited as follows:

      1. The saturation activity of an encapsulated fueled experiment is limited to 0.290 Ci of I-131 unless the criteria of 3.8(d)(iii)(2) is met.

      2. The saturation activity of an encapsulated fueled experiment with the primary encapsulation fully submerged underwater in the reactor pool is limited to 1.57 Ci of I-131.

   iv. Vented fueled experiments shall meet the following:

      1. Only fission gases and halogen releases are allowed.

      2. Released activity per calendar day from fission gases at a rate that does not exceed[(1)]:

i. 9.75 Ci per day from 1 to 93 days in a calendar year as depicted in Figure 3.8-1.

~~i.~~ii. 910 Ci divided by 93 days to 365 days in a calendar year as depicted in Figure 3.8-1.

1. Experiment exhaust is filtered for particulates and halogens.(2)
2. Experiment exhaust is decayed for a minimum of 100 minutes.
3. Experiment exhaust flow rate is monitored.
4. Experiment exhaust gas is monitored for radioactivity.
5. Halogens in the stack particulate radiation monitoring channel are monitored.

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) The released activity is controlled to ensure that the 10 CFR 20 annual dose constraint for all airborne effluent of 0.01 rem is met.

~~e.~~ (2) At flow rates specified for the vented fueled experiment, removal efficiency for the filter train shall be 0.99 or greater. Removal efficiency for each filter in the filter train shall be 0.95 or greater. Filter efficiencies shall be maintained for the entire duration that the filter is in use.~~Specification 3.7 pertaining to reactor experiments shall be met.~~

~~d. Specification 6.5 pertaining to the review of experiments shall be met. Each~~

~~type of fueled experiment shall be classified as a new (untried) experiment with a documented review. The documented review shall include the following items:~~

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

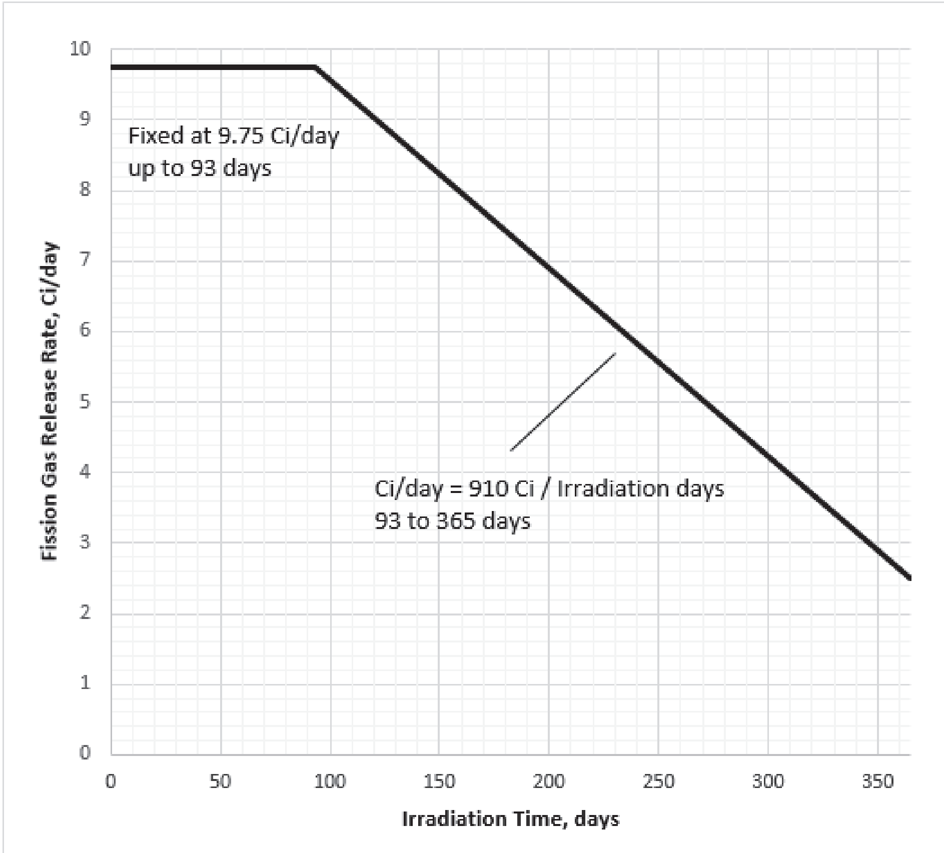
~~ii. Limiting the thermal power generated from the fissile material to ensure that the surface temperature of the experiment does not exceed the saturation temperature of the reactor pool water.~~

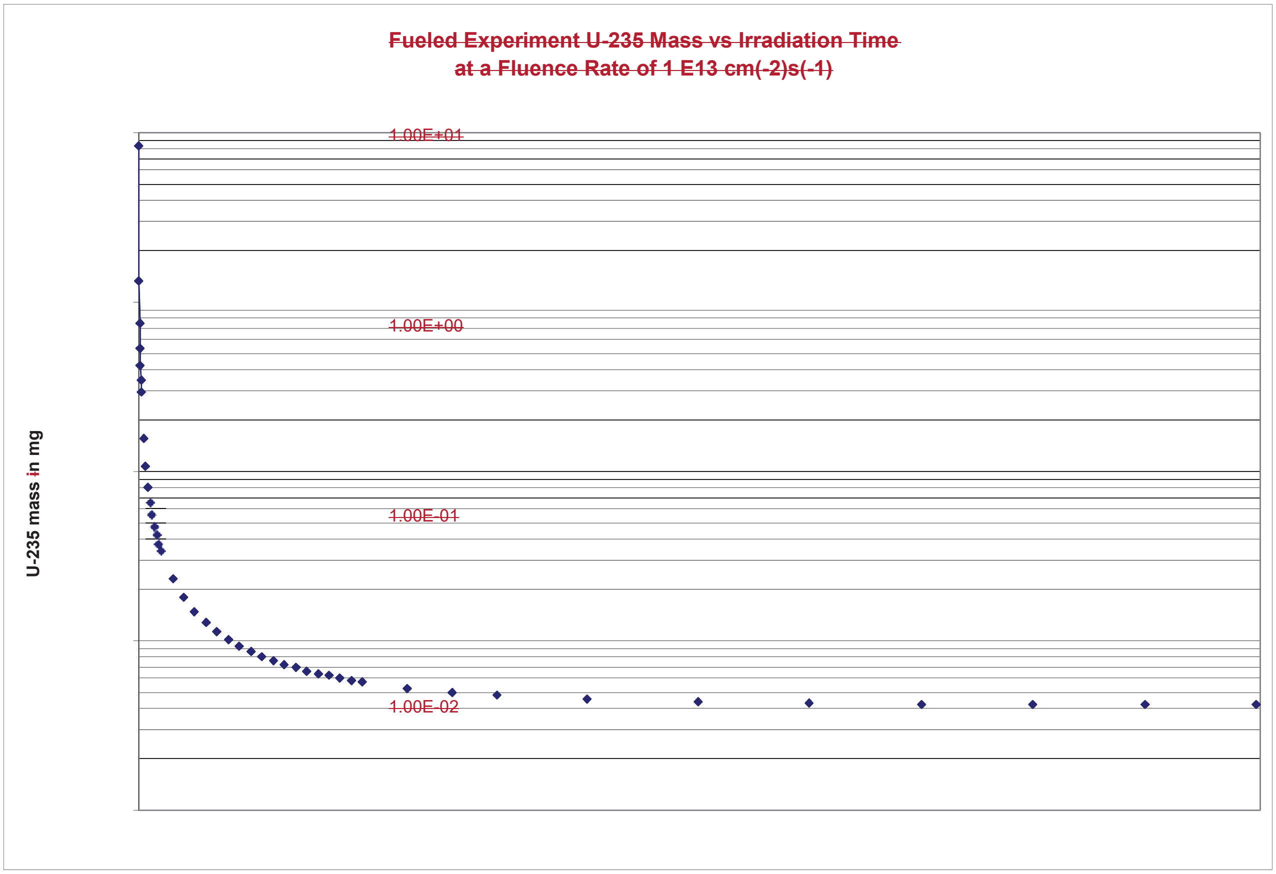
~~iii. Radiation monitoring for detection of released fission products.~~

~~iii. Design criteria related to meeting conditions given in Specifications 3.2 and 3.7.~~

~~Credible failure of any fueled experiment shall not result in releases or exposures in excess of 10 percent of the annual limits established in 10 CFR Part 20.~~

Figure 3.8-1: Vented Fueled Experiment Fission Gas Release Rate





~~**Figure 3.8-1**~~

~~**NOTE: The mass at 500 hours may be used for periods up to 1 y (8760 hours).**~~

~~**Table 3.8-1: Data for Fueled Experiments at a Fluence Rate of 1E13 cm^{-2}s^{-1}**~~

~~Irradiation Time, s~~	~~U-235 Mass mg~~	~~Mass-Fluence mg cm(-2)~~	~~Fission Rate f/s~~	~~Power milliwatts~~
~~6.00E+01~~	~~8.34E+00~~	~~5.00E+15~~	~~1.25E+11~~	~~4.01E+03~~
~~1.20E+02~~	~~4.75E+00~~	~~5.70E+15~~	~~7.12E+10~~	~~2.28E+03~~
~~1.80E+02~~	~~3.44E+00~~	~~6.19E+15~~	~~5.16E+10~~	~~1.65E+03~~
~~3.00E+02~~	~~2.30E+00~~	~~6.90E+15~~	~~3.45E+10~~	~~1.10E+03~~
~~6.00E+02~~	~~1.33E+00~~	~~7.98E+15~~	~~1.99E+10~~	~~6.39E+02~~
~~1.20E+03~~	~~7.55E-01~~	~~9.06E+15~~	~~1.13E+10~~	~~3.63E+02~~
~~1.80E+03~~	~~5.36E-01~~	~~9.65E+15~~	~~8.04E+09~~	~~2.57E+02~~
~~2.40E+03~~	~~4.18E-01~~	~~1.00E+16~~	~~6.27E+09~~	~~2.01E+02~~
~~3.00E+03~~	~~3.43E-01~~	~~1.03E+16~~	~~5.14E+09~~	~~1.65E+02~~
~~3.60E+03~~	~~2.92E-01~~	~~1.05E+16~~	~~4.38E+09~~	~~1.40E+02~~
~~7.20E+03~~	~~1.57E-01~~	~~1.13E+16~~	~~2.35E+09~~	~~7.54E+01~~
~~1.08E+04~~	~~1.08E-01~~	~~1.17E+16~~	~~1.62E+09~~	~~5.19E+01~~
~~1.44E+04~~	~~8.13E-02~~	~~1.17E+16~~	~~1.22E+09~~	~~3.90E+01~~
~~1.80E+04~~	~~6.55E-02~~	~~1.18E+16~~	~~9.82E+08~~	~~3.15E+01~~

~~2.16E+04~~	~~5.49E-02~~	~~1.19E+16~~	~~8.23E+08~~	~~2.64E+01~~
~~2.52E+04~~	~~4.74E-02~~	~~1.19E+16~~	~~7.11E+08~~	~~2.28E+01~~
~~2.88E+04~~	~~4.18E-02~~	~~1.20E+16~~	~~6.27E+08~~	~~2.01E+01~~
~~3.24E+04~~	~~3.74E-02~~	~~1.21E+16~~	~~5.61E+08~~	~~1.80E+01~~
~~3.60E+04~~	~~3.39E-02~~	~~1.22E+16~~	~~5.08E+08~~	~~1.63E+01~~
~~7.20E+04~~	~~1.81E-02~~	~~1.30E+16~~	~~2.71E+08~~	~~8.69E+00~~
~~1.08E+05~~	~~1.28E-02~~	~~1.38E+16~~	~~1.92E+08~~	~~6.15E+00~~
~~1.44E+05~~	~~1.02E-02~~	~~1.47E+16~~	~~1.53E+08~~	~~4.90E+00~~
~~1.80E+05~~	~~8.67E-03~~	~~1.56E+16~~	~~1.30E+08~~	~~4.16E+00~~
~~2.16E+05~~	~~7.66E-03~~	~~1.65E+16~~	~~1.15E+08~~	~~3.68E+00~~
~~2.52E+05~~	~~6.95E-03~~	~~1.75E+16~~	~~1.04E+08~~	~~3.34E+00~~
~~2.88E+05~~	~~6.42E-03~~	~~1.85E+16~~	~~9.62E+07~~	~~3.08E+00~~
~~3.24E+05~~	~~6.03E-03~~	~~1.95E+16~~	~~9.04E+07~~	~~2.90E+00~~
~~3.60E+05~~	~~5.72E-03~~	~~2.06E+16~~	~~8.57E+07~~	~~2.75E+00~~
~~3.96E+05~~	~~5.27E-03~~	~~2.09E+16~~	~~7.90E+07~~	~~2.53E+00~~
~~4.32E+05~~	~~4.97E-03~~	~~2.15E+16~~	~~7.45E+07~~	~~2.39E+00~~
~~4.68E+05~~	~~4.77E-03~~	~~2.23E+16~~	~~7.15E+07~~	~~2.29E+00~~
~~7.20E+05~~	~~4.51E-03~~	~~3.25E+16~~	~~6.76E+07~~	~~2.17E+00~~
~~1.08E+06~~	~~4.27E-03~~	~~4.61E+16~~	~~6.40E+07~~	~~2.05E+00~~
~~1.44E+06~~	~~4.21E-03~~	~~6.06E+16~~	~~6.31E+07~~	~~2.02E+00~~
~~1.80E+06~~	~~4.19E-03~~	~~7.54E+16~~	~~6.28E+07~~	~~2.01E+00~~
~~2.16E+06~~	~~4.19E-03~~	~~9.05E+16~~	~~6.28E+07~~	~~2.01E+00~~
~~4.32E+06~~	~~4.19E-03~~	~~1.81E+17~~	~~6.28E+07~~	~~2.01E+00~~
~~4.32E+06~~	~~4.19E-03~~	~~1.81E+17~~	~~6.28E+07~~	~~2.01E+00~~
~~1.73E+07~~	~~4.19E-03~~	~~7.24E+17~~	~~6.28E+07~~	~~2.01E+00~~
~~3.15E+07~~	~~4.19E-03~~	~~1.32E+18~~	~~6.28E+07~~	~~2.01E+00~~

Bases

1) Specification 3.8.a requires all specifications pertaining to experiment reactivity given in TS 3.2 be satisfied 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 ensuring releases are monitored and filtered if an abnormal release is detected.

3) Specification 3.8.c requires all specifications pertaining to limitations on experiments given in TS 3.7 be satisfied 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 lists the fissionable materials allowed in fueled experiments.

   c. Specification 3.8.d.iii gives the saturation activity of I-131 for encapsulated experiments that result in a TEDE of 0.01 rem to public areas outside the reactor building if an accidental release were to occur. An encapsulated experiment that is irradiated fully submerged underwater in the reactor pool takes credit for halogen retention in water in the event of encapsulation failure. Fission gas radionuclides of Kr include 83m, 85m, 85, 87, 88, and 89. Fission gas radionuclides of Xe include 131m, 133m, 133, 135m, 135, 137, and 138. TEDE doses are calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19.

   d. Specification 3.8.d.iv limits fission gas release and provides experimental controls that result in a TEDE of 0.01 rem or less in public areas outside the reactor building from vented fueled experiments. Fission gas radionuclides of Kr include 83m, 85m, 85, 87, 88, and 89. Fission gas radionuclides of Xe include 131m, 133m, 133, 135m, 135, 137, and 138. TEDE doses are calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19.

   e. For Specifications 3.8 d.iii and 3.8 d.iv, radiation doses were calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19. Footnote (1) to TS 3.8.d.iv.3 specifies the required filter removal efficiency. Meeting Specification 3.8.d.iii and 3.8 d.iv gives a TEDE less than 1 rem and the Total Organ Dose-Equivalent to the thyroid (TODE) less than 25 rem to occupants inside the reactor building.

   ~~a.~~f. Specification 3.8.e requires all specifications pertaining to criticality control given in TS 5.3 be satisfied thus ensuring that fueled experiments are stored in sub-critical configurations.

g. 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.

~~NUREG 1537 provides guidelines for the format and content of non-power reactor licensing. Guidelines on operating conditions and accident analysis for fueled experiments are given in NUREG 1537. These guidelines include (1) actuation of engineered safety features (ESF) to prevent or mitigate the consequences of damage to fission product barriers caused by overpower or loss of cooling events, (2) use of ESF to control of radioactive material released by accidents, (3) radiation monitoring of fission product effluent and accident releases, (4) accidental analysis for loss of cooling or other experimental malfunction resulting in liquefaction or volatilization of fissile materials, (5) accident analysis for catastrophic failure of the experiment in the reactor pool or air, (6) accident analysis for insertion of excess reactivity leading to fuel melting, and (7) emergency plan activation and classification.~~

~~The limitations given in Specification 3.8 ensure that (1) fueled experiments performed in experimental facilities at the reactor prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure, (2) radiation doses to occupational personnel and the public and radioactive material releases are ALARA, (3) adequate radiation monitoring is in place, and (4) in the event of failure of a fueled experiment with the subsequent release of radioactive material, the resulting dose to personnel and the public at any location are well within limits set in 10 CFR Part 20.~~

~~Specification 3.8.e ensures that each type of fueled experiment is reviewed, approved, and documented as required by Specification 6.5. This includes (1) meeting applicable limitations on experiments given in Specifications 3.2 and 3.7, (2) limiting the amount of fissile material to ensure that experimental reactivity conditions are met and that radiation doses are well within 10 CFR Part 20 limits following maximum fission product release from a failed experiment, and (3) limiting the thermal power generated from the fissile material to ensure that the surface temperature of the experiment does not exceed the saturation temperature of the reactor pool water.~~

4.4 Radiation Monitoring Equipment

Applicability

This specification applies to the surveillance requirements for the area and stack effluent radiation monitoring equipment.

Objective

The objective is to assure that the radiation monitoring equipment is operable.

Specification

a. The area and stack monitoring systems shall be calibrated annually but at intervals not to exceed fifteen (15) months.

b. The setpoints shall be verified monthly, but shall not exceed (6) weeks.

c. Vented fueled experiment exhaust gas radiation and flow monitors shall be calibrated prior to initial operation of the experiment and annually, not to exceed 15 months, thereafter for as long as the experiment is in operation.

d. Filter replacements for vented fueled experiments shall be biennial, but shall not exceed 30 months, and shall have a removal efficiency for iodine adsorption of 0.95 or greater at the specified flow rates used.

e. Leak testing of the filter housing(s) used for vented fueled experiments is performed after new filters are installed and prior to initial use; e.g. by pressure testing.

~~b. weekly, but at intervals not to exceed ten~~
~~c. (10) days.~~

Bases

These systems provide continuous radiation monitoring of the Reactor Building with a check of readings performed prior to and during reactor operations.

The monthly verification of the setpoints in conjunction with the annual calibration is adequate to identify long term variations in the system operating characteristics.

Filter replacement of 2 years to 2.5 years (30 months), is consistent with TS 4.5 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.

~~Therefore, the weekly verification of the setpoints in conjunction with the annual calibration is adequate to identify long term variations in the system operating characteristics.~~

ATTACHMENT 3:
Technical Specifications Amendment 19

Appendix A

Technical Specifications for the North Carolina State University PULSTAR Reactor

Facility License No. R-120

Docket No. 50-297

Amendment No. 19

Date: April 18, 2022

TABLE OF CONTENTS

FIGURES

TABLES

1.0 INTRODUCTION

1.1 Purpose

These Technical Specifications provide limits within which operation of the reactor will assure the health and safety of the public, the environment and on-site personnel. Areas addressed are Definitions, Safety Limits (SL), Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements, Design Features, and Administrative Controls.

Included in this document are the "Bases" for the Technical Specifications. The bases provide the technical support for the individual technical specification and are included for information purposes only. The bases are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.2 Definitions

1.2.1 Channel: A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

1.2.2 Channel Calibration: A channel calibration is an adjustment of the channel, such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include a Channel Test.

1.2.3 Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.

1.2.4 Channel Test: A channel test is the introduction of a signal into the channel for verification that it is operable.

1.2.5 Cold Critical: The condition of the reactor when it is critical, with negligible xenon, and the fuel and bulk water are both at an isothermal temperature of 70°F.

1.2.6 Confinement: Confinement means a closure on the overall facility that controls the movement of air into and out of the facility through a controlled path.

1.2.7 Control Rod: A control rod is a neutron absorbing blade having an in-line drive which is magnetically coupled and has SCRAM capability.

1.2.8 Excess Reactivity: Excess reactivity is that amount of reactivity that would exist if all control rods (and Shim Rod) were fully withdrawn from the point where the reactor is exactly critical ($k_{eff}=1$).

1.2.9 Experiment: Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beam tube or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be a part of their design. Specific categories of experiments include:

a. **Tried Experiment**: Tried experiments are those experiments that have been previously performed in this reactor. Specifically, a tried experiment has similar size, shape, composition and location of an experiment previously approved and performed in the reactor.

b. **Secured Experiment**: A secured experiment is any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

c. **Non-Secured Experiment**: A non-secured experiment is an experiment that does not meet the criteria for being a "secured" experiment.

d. **Movable Experiment**: A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

e. **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.

1.2.10 **Experimental Facilities**: Experimental facilities are facilities used to perform experiments. They include beam tubes, thermal columns, void tanks, pneumatic transfer systems, in-core facilities at single-assembly positions, out-of-core irradiation facilities, and the bulk irradiation facility.

1.2.11 **Limiting Condition for Operation**: Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility (10 CFR 50.36).

1.2.12 **Limiting Safety System Setting**: Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded (10 CFR 50.36).

1.2.13 **Measured Value**: The measured value is the value of a parameter as it appears on the output of a channel.

1.2.14 **Operable**: Operable means a component or system is capable of performing its intended function.

1.2.15 **Operating**: Operating means a component or system is performing its intended function.

1.2.16 **pcm**: A unit of reactivity that is the abbreviation for "percent millirho" and is equal to 10^{-5} $\Delta k/k$ reactivity. For example, 1000 pcm is equal to 1.0% $\Delta k/k$.

1.2.17 Reactor Building: The Reactor Building includes the Reactor Bay, Control Room and Ventilation Room, the Mechanical Equipment Room (MER), and the Primary Piping Vault (PPV). The Nuclear Regulatory Commission R-120 license applies to the areas in the Reactor Building and the Waste Tank Vault.

1.2.18 Reactor Operation: Reactor operation is any condition when the reactor is not secured or shutdown.

1.2.19 Reactor Operator: A reactor operator (RO) is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the facility.

1.2.20 Reactor Operator Assistant (ROA): An individual who has been certified by successful completion of an in-house training program to assist the licensed reactor operator during reactor operation.

1.2.21 Reactor Safety System: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.2.22 Reactor Secured: The reactor is secured when:

a. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection, **or**

b. The following conditions exist:

   i. All scrammable neutron absorbing control rods are fully inserted, **and**

   ii. The reactor key switch is in the OFF position and the key is removed from the lock, **and**

   iii. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, **and**

   iv. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding one dollar (730 pcm).

1.2.23 Reactor Shutdown: That subcritical condition of the reactor where the absolute value of the negative reactivity of the core is equal to or greater than the shutdown margin.

1.2.24 Reportable Event: A Reportable Event is any of the following:

a. Violation of a Safety Limit.

b. Release of radioactivity from the site above allowed limits.

c. Operation with actual Safety System Settings (SSS) for required systems less conservative than the Limiting Safety System Settings (LSSS) specified in these specifications.

d. Operation in violation of Limiting Conditions for Operation (LCO) established in these Technical Specifications.

e. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdown. (For components or systems other than those required by these Technical Specifications, the failure of the extra component or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

f. An unanticipated or uncontrolled change in reactivity greater than one dollar (730 pcm). Reactor trips resulting from a known cause are excluded.

g. Abnormal or significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks), which could result in exceeding radiological limits for personnel or environment, or both, as prescribed in the facility Emergency Plan.

h. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence of an unsafe condition with regard to reactor operations.

1.2.25 Safety Limit: Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity (10 CFR 50.36).

1.2.26 Shim Rod: A shim rod is a neutron absorbing rod having an in-line drive which is mechanically, rather than magnetically, coupled and does not have a SCRAM capability.

1.2.27 Senior Reactor Operator: A senior reactor operator (SRO) is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the facility and to direct the activities of licensed reactor operators.

1.2.28 Shutdown Margin: Shutdown margin means the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition with the most reactive scrammable rod fully withdrawn, the non-scrammable rod (Shim rod) fully withdrawn, and experiments considered at their most reactive condition, and finally, that the reactor will remain subcritical without further operator action.

1.2.29 Total Nuclear Peaking Factor: The factor obtained by multiplying the measured local radial and axial neutron fluence peaking factors.

1.2.30 True Value: The true value is the actual value of a parameter.

1.2.31 University Management: University Management is the Chancellor or Office of the Chancellor other University Administrator(s) having authority designated by the Chancellor or as specified in University policies.

1.2.32 Unscheduled Shutdown: An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation not including shutdowns that occur during testing or check-out operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits (SL)

2.1.1 Safety Limits for Forced Convection Flow

Applicability

This specification applies to the interrelated variables associated with the core thermal and hydraulic performance with forced convection flow. These interrelated variables are:

P	Reactor Thermal Power
W	Reactor Coolant Flow Rate
H	Height of Water Above the Top of the Core
$\mathbf{T}_{inlet}$	Reactor Coolant Inlet Temperature

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of forced convection flow, the Safety Limit shall be as follows:

a. The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 2.1-1 under any operating conditions. The limits are considered exceeded if the point defined by the true values of P and W is at any time outside the operating envelope shown in Figure 2.1-1.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

Bases

Above 80 percent of the full core flow of 500 gpm in the region of full power operation, the criterion used to establish the Safety Limit was no bulk boiling at the outlet of any coolant channel. This was found to be far more limiting than the criterion of a minimum allowable burnout heat flux ratio of 2.0. The analysis is given in the SAR Appendix 3B.

In the region below 80 percent of full core flow, where, under a loss of flow transient at power the flow coasts down to zero, reverses, and then establishes natural convection, the criterion for selecting a Safety Limit is taken as a fuel cladding temperature. The analysis of a loss of flow transient is presented in Appendix 3B of the SAR. For initial conditions of full flow and an operating power of 1.4 MWt, the maximum clad temperature reached under the conservative assumptions of the analysis was 273°F which is well below the temperature at which fuel clad damage could possibly occur. The Safety Limit shown in Figure 2.1-1 for flow less than 80 percent of full flow is the steady state power corresponding to the maximum fuel clad temperature of 273°F with natural convection flow, namely, 1.4 MWt.

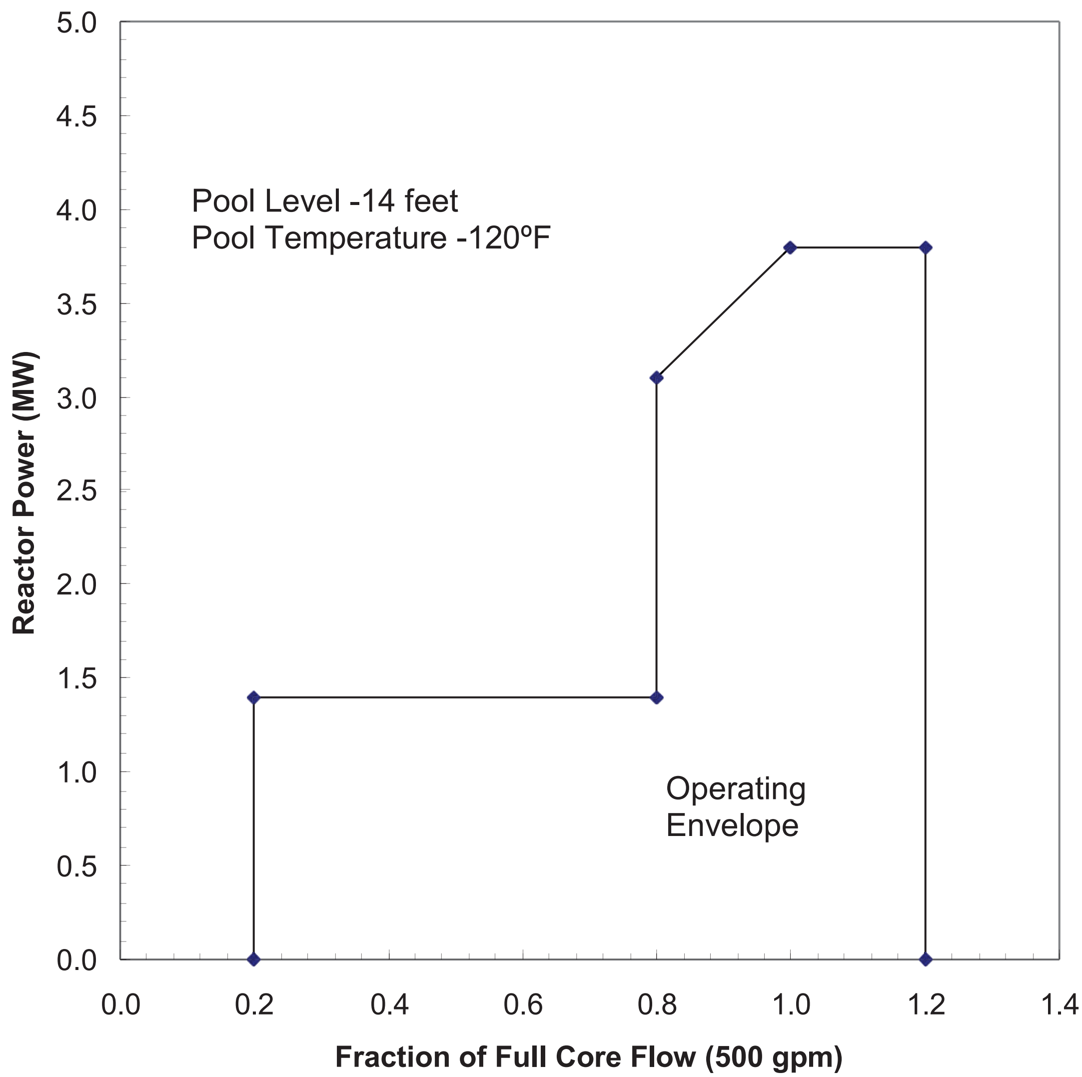


Figure 2.1-1: Power-Flow Safety Limit Curve

2.1.2 Safety Limits for Natural Convection Flow

Applicability

This specification applies to the interrelated variables associated with the core thermal and hydraulic performance with natural convection flow. These interrelated variables are:

P	Reactor Thermal Power
H	Height of Water Above the Top of the Core
$\mathbf{T}_{inlet}$	Reactor Coolant Inlet Temperature

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of natural convection flow, the Safety Limit shall be as follows:

a. The true value of reactor thermal power (P) shall not exceed 1.4 MWt.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

Bases

The criterion for establishing a Safety Limit with natural convection flow is established as the fuel clad temperature. This is consistent with Figure 2.1-1 for forced convection flow during a transient. The analysis of natural convection flow given in Appendix 3B and 3C of the SAR shows that at 1.4 MWt the maximum fuel clad temperature is 273°F which is well below the temperature at which fuel clad damage could occur. The flow with natural convection at this power is 98 gpm. This flow is based on data from natural convection tests with fuel assemblies of the same design performed in the prototype PULSTAR Reactor, as referenced in Section 3 of the SAR.

2.2 Limiting Safety System Settings

2.2.1 Limiting Safety System Settings (LSSS) for Forced Convection Flow

Applicability

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), height of water above the top of the core (H), and pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specification

Under the condition of forced convection flow, the Limiting Safety System Settings shall be as follows:

P	1.3 MWt (max.)
W	450 gpm (min.)
H	14 feet, 2 inches (min.)
T	117°F

Bases

The Limiting Safety System Settings that are given in the Specification 2.2.1 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded during the most limiting anticipated transient (loss of flow). The safety margin that is provided between the Limiting Safety System Settings and the Safety Limits also allows for the most adverse combination of instrument uncertainties associated with measuring the observable parameters. These instrument uncertainties include a flow variation of ten percent, a pool level variation of two inches and a power level variation of seven percent.

The analysis presented in Section 3 of the SAR of a loss of flow transient indicates that if the interrelated variables were at their LSSS, as specified in 2.2.1 above, at the initiation of the transient, the Safety Limits specified in 2.1.1 would not be exceeded.

2.2.2. Limiting Safety System Settings (LSSS) for Natural Convection Flow

Applicability

This specification applies to the setpoints for the safety channel monitoring reactor thermal power (P), the height of water above the core (H), and the pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specifications

Under the condition of natural convection flow, the Limiting Safety System Settings shall be as follows:

P	250 kWt (max.)
H	14 feet, 2 inches (min.)
T	117°F

Bases

The Limiting Safety System Settings that are given in Specification 2.2.2 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded. The specifications given above assure that an adequate safety margin exists between the LSSS and the SL for natural convection. The safety margin on reactor thermal power was chosen with the additional consideration related to bulk boiling at the outlet of the hot channel. This criterion is not related to fuel clad damage (for these relatively low power levels) which was the criterion used in establishing the Safety Limits (see Specification 2.1.2). It is desirable to minimize to the greatest extent practical, N-16 dose at the pool surface which might be aided by steam bubble rise during up-flow in natural convection. Analysis of coolant bulk boiling given in SAR, Section 3, indicates that the large safety margin on reactor thermal power assumed in Specification 2.2.2 above will satisfy this additional criterion of no bulk boiling in any channel.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Configuration

Applicability

This specification applies to the reactor core configuration during forced convection or natural convection flow operations.

Objective

The objective is to assure that the reactor will be operated within the bounds of established Safety Limits.

Specification

The reactor shall not be operated unless the following conditions exist:

a. A maximum of twenty-five fuel assemblies.

b. A maximum of ten reflector assemblies of either graphite or beryllium or a combination of these located on the core periphery.

c. Unoccupied grid plate penetrations plugged.

d. A minimum of four control rod guides are in place.

e. The maximum worth of a single fuel assembly shall not exceed 1590 pcm.

f. The total nuclear peaking factor in any fuel assembly shall not exceed 2.92.

Bases

Specifications 3.1.a through 3.1.d require that the core be configured such that there is no bypass cooling flow around the fuel through the grid plate.

Specification 3.1.e provides assurances that a fuel loading accident will not result in a Safety Limit to be exceeded as discussed in SAR Section 13.2.2.1.

Specification 3.1.f provides assurances that core hot channel power are bounded by the SAR assumptions in Appendix 3-B.

3.2 Reactivity

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods, shim rod and experiments.

Objective

The objective is to assure that the reactor can be shutdown at all times and that the Safety Limits will not be exceeded.

Specifications

The reactor shall not be operated unless the following conditions exist:

a. The shutdown margin, with the highest worth scrammable control rod fully withdrawn, with the shim rod fully withdrawn, and with experiments at their most reactive condition, relative to the cold critical condition, is greater than 400 pcm.

b. The excess reactivity is not greater than 3970 pcm.

c. The drop time of each control rod is not greater than 1.0 second.

d. The rate of reactivity insertion of the control rods is not greater than 100 pcm per second (critical region only).

e. The absolute reactivity worth of experiments or their rate of reactivity change shall not exceed the values indicated in Table 3.2-1.

f. The sum of the absolute values of the reactivity worths of all experiments shall not be greater than 2890 pcm.

Table 3.2-1: Reactivity Limits for Experiments

Experiment	Limit
Movable	300 pcm or 100 pcm/sec, whichever is more limiting
Non-secured	1000 pcm
Secured	1590 pcm

Bases

The shutdown margin required by Specification 3.2.a assures that the reactor can be shut down from any operating condition and will remain shutdown after cool down and xenon decay, even if the highest worth scrammable rod should be in the fully withdrawn position. Refer to Section 3.1.2.1.

The upper limit on excess reactivity ensures that an adequate shutdown margin is maintained.

The rod drop time required by Specification 3.2.c assures that the Safety Limit will not be exceeded during the flow reversal which occurs upon loss of forced convection coolant flow. The rise in fuel temperature due to heat storage is partially controlled by the reactivity insertion associated with the SCRAM. The analysis of this transient is based upon this SCRAM reactivity insertion taking the form of a ramp function of two second duration. This analysis is found in SAR Section 3.2.4 and Appendix 3B. The rod drop time is the time interval measured between the instant of a test signal input to the SCRAM Logic Unit and the instant of the rod seated signal.

The maximum rate of reactivity insertion by the control rods which is allowed by Specification 3.2.d assures that the Safety Limit will not be exceeded during a startup accident due to a continuous linear reactivity insertion. Refer to SAR Section 13.

Experiments affecting the reactivity condition of the reactor are commonly categorized by the sign of the reactivity effect produced by insertion of the experiment. An experiment having a large reactivity effect of either sign can also produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculations and the calibration of Safety Channels.

The Specification 3.2.e is intended to prevent inadvertent reactivity changes during reactor operation caused by the insertion or removal of an experiment. It further provides assurance that the failure of a single experiment will not result in a reactivity insertion which could cause the Safety Limit to be exceeded. Analyses indicate that the inadvertent reactivity insertion of these magnitudes will not result in consequences greater than those analyzed in the SAR Sections 3 and 13.

The total limit on reactivity associated with experiments ensures that an adequate shutdown margin is maintained.

3.3 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to require the minimum number of reactor safety system channels which must be operable in order to assure that the Safety Limits are not exceeded.

Specification

The reactor shall not be operated unless the reactor safety system channels described in Table 3.3-1 are operable.

Table 3.3-1: Required Safety and Safety Related Channels

	Measuring Channel	Function
a.	Startup Power Level[(1)]	Inhibits Control Rod withdrawal when neutron count is ≤2 cps
b.	Safety Power Level	SCRAM at ≤1.3 MW (LSSS) Enable for Flow/Flapper SCRAMs at ≤250 kW (LSSS)
c.	Linear Power Level	SCRAM at ≤1.3 MW (LSSS)
d.	Log N Power Level	Enable for Flow/Flapper SCRAMs at ≤250 kW (LSSS)
e.	Flow Monitoring[(2)]	SCRAM when flapper not closed and Flow/Flapper SCRAMs are enabled
f.	Primary Coolant Flow[(2)]	SCRAM at ≥450 gpm (LSSS) when Flow/Flapper SCRAMs are enabled
g.	Pool Water Temperature Monitoring Switch	ALARM at ≤117°F
h.	Pool Water Temperature Measuring Channel	SCRAM at ≤117°F (LSSS)
i.	Pool Water Level	SCRAM at ≥14 feet 2 inches
j.	Manual SCRAM Button	SCRAM
k.	Reactor Key Switch	SCRAM
l.	Over-the-Pool Radiation Monitor[(3)]	Alarm (100 mR/hr)

(1) Required only for reactor startup when power level is less than 4 watts.

(2) Either the Flapper SCRAM or the Flow SCRAM may be bypassed during maintenance testing and/or performance of a startup checklist in order to verify each SCRAM is independently operable. The reactor must be shutdown in order to use these bypasses.

(3) May be bypassed for less than two minutes during the return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

Bases

The Startup Channel inhibit function assures the required startup neutron source is sufficient and in its proper location for the reactor startup, such that a minimum source multiplication count rate level is being detected to assure adequate information is available to the operator.

The reactor power level SCRAMs provide the redundant protection channels to assure that, if a condition should develop which would tend to cause the reactor to operate at an abnormally high power level, an immediate automatic protective action will occur to prevent exceeding the Safety Limit.

The primary coolant flow SCRAMs provide redundant channels to assure when the reactor is at power levels which require forced flow cooling that, if sufficient flow is not present, an immediate automatic shutdown of the reactor will occur to prevent exceeding a Safety Limit. The Log N Power Channel is included in this section since it is one of the two channels which enables the two flow SCRAMs when the reactor is above 250 kW (LSSS).

The pool water temperature channel provides for shutdown of the reactor and prevents exceeding the Safety Limit due to high pool water temperature.

The pool water level channel together with the Over-the-Pool (Bridge) radiation monitor, provides two diverse channels for shutdown of the reactor and prevents exceeding the Safety Limit due to insufficient pool height.

To prevent unnecessary initiation of the evacuation and confinement systems during the return of the pneumatic capsule from the core to the unloading station or during the removal of experiments from the reactor pool, the Over-the-Pool monitor may be bypassed for the specified time interval.

The manual SCRAM button and the Reactor Key switch provide two manual SCRAM methods to the reactor operator if unsafe or abnormal conditions should occur.

3.4 Reactor Instrumentation

Applicability

This specification applies to the instrumentation that shall be available to the reactor operator to support the safe operation of the reactor, but are not considered reactor safety systems.

Objective

The objective is to require that sufficient information be available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the following are operable:

a. N-16 Power Measuring Channel when reactor power is greater than 500 kW

b. Control Rod Position Indications for each control rod and the Shim Rod

c. Differential pressure gauge for "Bay with Respect to Atmosphere"

Bases

The N-16 Channel provides the necessary power level information to allow adjustment of Safety and Linear Power Channels.

Control rod position indications give the operator information on rod height necessary to verify shutdown margin.

The differential pressure gauge provides the pressure difference between the Reactor Bay and the outside ambient and confirms air flow in the ventilation stream for both normal and confinement modes.

3.5 Radiation Monitoring Equipment

Applicability

This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective

To assure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specification

The reactor shall not be operated nor shall irradiated fuel or irradiated fueled experiments be moved within the reactor building unless the radiation monitoring equipment listed below and in Table 3.5-1 are operable.(1)(2)(3)(7)

a. Three fixed area monitors operating in the Reactor Building with their setpoints as listed in Table 3.5-1.(1)(3)(4)

b. Stack Gas and Stack Particulate building exhaust monitors continuously sampling air in the facility exhaust stack with their setpoints as listed in Table 3.5-1.(1)(3)(4)

c. The Radiation Rack Recorder.(5)

Table 3.5-1: Required Radiation Area Monitors

Monitor	Alert Setpoint	Alarm Setpoint
Control Room	≤ 2 mR/hr	≤ 5 mR/hr
Over-the-Pool	≤ 5 mR/hr	≤ 100 mR/hr
West Wall	≤ 5 mR/hr	≤ 100 mR/hr
Stack Gas	Sum of count rate from 1.1×10^{-5} µCi/ml of Ar-41 and 1.3×10^{-4} µCi/ml of Fission Gas (6)(9)	Lower count rate from 9.4×10^{-5} µCi/ml of Ar-41 or 5.4×10^{-4} µCi/ml of Fission Gas (6)(9)
Stack Particulate	Lower count rate from 4.4×10^{-7} µCi/ml of Co-60 or 1.0×10^{-6} µCi/ml of Fission Halogens (6)(9)	Lower count rate from 6.6×10^{-7} µCi/ml of Co-60 or 1.7×10^{-6} µCi/ml of Fission Halogens (6)(9)

d. Vented fueled experiment exhaust gas radiation monitor continuously monitoring the experiment exhaust gas.(7)(8)

e. Vented fueled experiment flow rate monitor continuously monitoring the experiment exhaust gas flow.(7)(8)

(1) For periods of time, not to exceed ninety days, for maintenance to the radiation monitoring channel, the intent of this specification will be satisfied if one of the installed channels is replaced with a gamma-sensitive instrument which has its own alarm audible or observable in the control room. Refer to SAR Section 5.

(2) The Over-the-Pool Monitor may be bypassed for less than two minutes during return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

(3) Stack Gas and Particulate are based on the AEC quantities present in the ventilation flow stream as it exits the stack. Refer to SAR Section 10 for setpoint bases for the radiation monitoring equipment.

(4) May be bypassed for less than one minute immediately after starting the pneumatic blower system.

(5) During repair and/or maintenance of the recorder not to exceed 90 days, the specified area and effluent monitor readings shall be recorded manually at a nominal interval of 30 minutes when the reactor is not shutdown. Refer to SAR Section 5.

(6) Co-60 and Ar-41 are radionuclides of concern without a fueled experiment in progress.

(7) Monitors for vented fueled experiments are only required to be operable while the experiment is in operation.

(8) Vented fueled experiment exhaust gas radiation and flow monitor setpoints meet Specification 3.8.d.iv and isolate the experiment exhaust when exceeded.

(9) Fission gases and halogens are released from vented fueled experiments or fueled experiment accidents.

Bases

A continued evaluation of the radiation levels within the Reactor Building will be made to assure the safety of personnel. This is accomplished by the area monitoring system of the type described in Section 5 of the Safety Analysis Report (SAR).

Evaluation of the continued discharge air to the environment will be made using the information recorded from the stack particulate and stack gas monitors.

When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, the building will be automatically placed in confinement as described in SAR Section 5.

To prevent unnecessary initiation of the evacuation confinement system during the return of a pneumatic capsule from the core to the unloading station or during removal of experiments from the reactor pool, the Over-the-Pool Monitor may be bypassed during the specified time interval. Refer to SAR Section 5.

Stack gas and stack particulate setpoints are based on the Notification of Unusual Event Emergency Action Level (EAL) for Ar-41 and Co-60, respectively, during normal operation with no vented fueled experiments being performed.

Radiation dose for EAL are higher than those for fueled experiments. While fueled experiments are performed, the stack gas and stack particulate monitor setpoints are changed to meet setpoints listed in Table 3.5-1.

In addition, the exhaust fission gases and exhaust flow rate from vented fueled experiments are monitored in accordance with Specification 3.8. Upon reaching an alarm setpoint from the vented fueled experiment exhaust gas radiation or flow monitors, the vented fueled experiment exhaust is automatically isolated. Radiation monitor setpoints are analyzed as described in the documentation presented in the Fueled Experiment Analysis Report for TS Amendment 19.

3.6 Confinement and Main HVAC Systems

Applicability

This specification applies to the operation of the Reactor Building confinement and main HVAC systems.

Objective

The objective is to assure that the confinement system is in operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated, nor shall irradiated fuel be moved within the pool area, unless the following equipment is operable, and conditions met:

Table 3.6-1: Required Main HVAC and Confinement Conditions

	Equipment/Condition	Function
a.	All doors, except the Control Room, and basement corridor entrance: self-latching, self-closing, closed and locked.	To maintain reactor building negative differential pressure (dp).(1)
b.	Control room and basement corridor entrance door: self-latching, self-closing and closed.	To maintain reactor building negative differential pressure.(2)
c.	Reactor Building under a negative differential pressure of not less than 0.2" H_2O with the normal ventilation system or 0.1" H_2O with one confinement fan operating.	To maintain reactor building negative differential pressure with reference to outside ambient.(3)
d.	Confinement system	Operable(4)(5)(7)
e.	Evacuation system	Operable(6)

(1) Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided audible and visual indications are available for the reactor operator to verify door status. Refer to SAR Section 5.

(2) Doors may be opened for periods of less than five minutes for personnel and equipment transport between corridor area and Reactor Building. Refer to SAR Section 5.

(3) During an interval not to exceed 30 minutes after a loss of dp is identified with Main HVAC operating, reactor operation may continue while the loss of dp is investigated and corrected. Refer to SAR Section 5.

(4) Operability also demonstrated with an auxiliary power source.

(5) One filter train may be out of service for the purpose of maintenance, repair, and/or surveillance for a period of time not to exceed 45 days. During the period of time in which one filter train is out of service, the standby filter train shall be verified to be operable every 24 hours if the reactor is operating with the Reactor Building in normal ventilation.

(6) The public address system can serve temporarily for the Reactor Building evacuation system during short periods of maintenance.

(7) When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, listed in Table 3.6-1, the building will be automatically placed in confinement as described in SAR Section 5.

Bases

In the event of a fission product release, the confinement initiation system will secure the normal ventilation fans and close the normal inlet and exhaust dampers. In confinement mode, a confinement system fan will: maintain a negative pressure in the Reactor Building and insure in-leakage only; purge the air from the building at a greatly reduced and controlled flow through charcoal and absolute filters; and control the discharge of all air through a 100 foot stack onsite. Section 5 of the SAR describes the confinement system sequence of operation.

The allowance for operation under a temporary loss of dp when in normal ventilation is based on the requirement of having the confinement system operable and therefore ready to respond in the unlikely event of an airborne release.

3.7 Limitations of Experiments

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities. Fueled experiments must also meet the requirements of Specification 3.8.

Objective

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

Specification

The reactor shall not be operated unless the following conditions governing experiments exist:

a. All materials to be irradiated shall be either corrosion resistant or encapsulated within a corrosion resistant container to prevent interaction with reactor components or pool water. Corrosive materials, liquids, and gases shall be doubly encapsulated.

b. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2. Pressure buildup inside any container shall be limited to 200 psi.

c. Cooling shall be provided to prevent the surface temperature of an experiment to be irradiated from exceeding the saturation temperature of the reactor pool water.

d. Experimental apparatus, material or equipment to be inserted in the reactor shall be positioned so as to not cause shadowing of the nuclear instrumentation, interference with control rods, or other perturbations which may interfere with safe operation of the reactor.

e. Concerning the material content of experiments, the following will apply:

   i. No experiment will be performed unless the major constituent of the material to be irradiated is known and a reasonable effort has been made to identify trace elements and impurities whose activation may pose the dominant radiological hazard. When a reasonable effort does not give conclusive information, one or more short irradiations of small quantities of material may be performed in order to identify the activated products.

ii. Attempts will be made to identify and limit the quantities of elements having very large thermal neutron absorption cross sections, in order to quantify reactivity effects.

iii. Explosive material[(1)] shall not be allowed in the reactor. Experiments in which the material is considered to be potentially explosive, either while contained, or if it leaks from the container, shall be designed to maintain seal integrity even if detonated, to prevent damage to the reactor core or to the control rods or instrumentation and to prevent any change in reactivity.

iv. Each experiment will be evaluated with respect to radiation induced physical and/or chemical changes in the irradiated material, such as decomposition effects in polymers.

v. Experiments involving cryogenic liquids[(1)] within the biological shield, flammable[(1)], or highly toxic materials[(1)] require specific procedures for handling and shall be limited in quantity and approved as specified in Specification 6.2.3.

f. Credible failure of any experiment shall not result in releases or exposures in excess of the annual limits established in 10 CFR Part 20.

[(1)] Defined as follows (reference - *Handbook of Laboratory Safety* - Chemical Rubber Company, 4th Ed., 1995, unless otherwise noted):

Toxic: A substance that has the ability to cause damage to living tissue when inhaled, ingested, injected, or absorbed through the skin (*Safety in Academic Chemistry Laboratories* - The American Chemical Society, 1994).

Flammable: Having a flash point below 73°F and a boiling point below 100°F. The flash point is defined as the minimum temperature at which a liquid forms a vapor above its surface in sufficient concentrations that it may be ignited as determined by appropriate test procedures and apparatus as specified.

Explosive: Any chemical compound, mixture, or device, where the primary or common purpose of which is to function by explosion with substantially simultaneous release of gas and heat, the resultant pressure being capable of destructive effects. The term includes, but is not limited to, dynamite, black powder, pellet powder, initiating explosives, detonators, safety fuses, squibs, detonating cord, igniter cord, and igniters.

Cryogenic: A cryogenic liquid is considered to be a liquid with a normal boiling point below -238°F (reference - *National Bureau of Standards Handbook 44*).

Bases

Specifications 3.7.a, 3.7.b, 3.7.c, and 3.7.d are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure; and, serve as a guide for the review and approval of new and untried experiments.

Specification 3.7.e ensures that no physical or nuclear interferences compromise the safe operation of the reactor, specifically, an experiment having a large reactivity effect of either sign could produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculation and/or safety channels calibrations. Review of experiments using the specifications of Section 3 and Section 6 will ensure the insertion of experiments will not negate the considerations implicit in the Safety Limits and thereby violate license conditions.

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.iv.1.

d. Fissionable material(s) in a fueled experiment shall meet the following:

   i. Physical form shall be solid, powder, or liquid.

   ii. Limited to any mixture of U, Pu, and Np.

   iii. Encapsulated fueled experiments are limited as follows:

      1. The saturation activity of an encapsulated fueled experiment is limited to 0.290 Ci of I-131 unless the criteria of 3.8(d)(iii)(2) is met.

      2. The saturation activity of an encapsulated fueled experiment with the primary encapsulation fully submerged underwater in the reactor pool is limited to 1.57 Ci of I-131.

   iv. Vented fueled experiments shall meet the following:

      1. Only fission gases and halogen releases are allowed.

      2. Released activity per calendar day from fission gases at a rate that does not exceed[(1)]:

         i. 9.75 Ci per day from 1 to 93 days in a calendar year as depicted in Figure 3.8-1.

         ii. 910 Ci divided by 93 days to 365 days in a calendar year as depicted in Figure 3.8-1.

3. Experiment exhaust is filtered for particulates and halogens.[(2)]
4. Experiment exhaust is decayed for a minimum of 100 minutes.
5. Experiment exhaust flow rate is monitored.
6. Experiment exhaust gas is monitored for radioactivity.
7. Halogens in the stack particulate radiation monitoring channel are monitored.

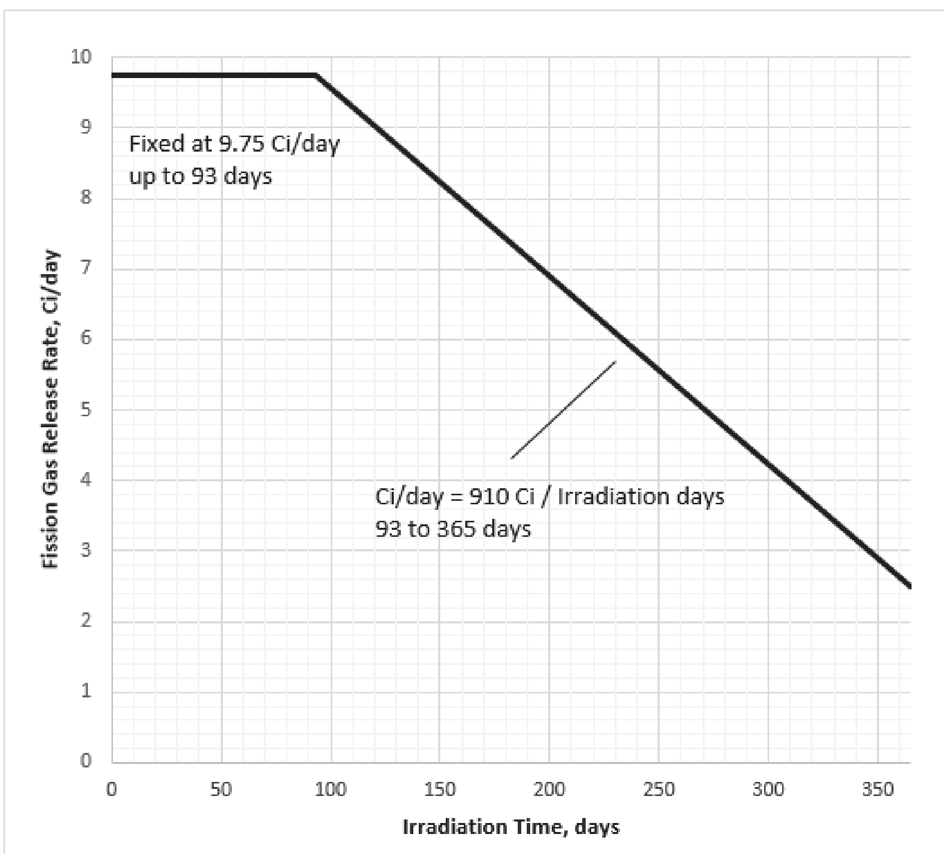
e. Specification 5.3 pertaining to criticality control for fueled experiments in storage shall be met.

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

(1) The released activity is controlled to ensure that the 10 CFR 20 annual dose constraint for all airborne effluent of 0.01 rem is met.

(2) At flow rates specified for the vented fueled experiment, removal efficiency for the filter train shall be 0.99 or greater. Removal efficiency for each filter in the filter train shall be 0.95 or greater. Filter efficiencies shall be maintained for the entire duration that the filter is in use.

Figure 3.8-1: Vented Fueled Experiment Fission Gas Release Rate



Bases

1) Specification 3.8.a requires all specifications pertaining to experiment reactivity given in TS 3.2 be satisfied 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 ensuring releases are monitored and filtered if an abnormal release is detected.
3) Specification 3.8.c requires all specifications pertaining to limitations on experiments given in TS 3.7 be satisfied 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 lists the fissionable materials allowed in fueled experiments.
    c. Specification 3.8.d.iii gives the saturation activity of I-131 for encapsulated experiments that result in a TEDE of 0.01 rem to public areas outside the reactor building if an accidental release were to occur. An encapsulated experiment that is irradiated fully submerged underwater in the reactor pool takes credit for halogen retention in water in the event of encapsulation failure. Fission gas radionuclides of Kr include 83m, 85m, 85, 87, 88, and 89. Fission gas radionuclides of Xe include 131m, 133m, 133, 135m, 135, 137, and 138. TEDE doses are calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19.
    d. Specification 3.8.d.iv limits fission gas release and provides experimental controls that result in a TEDE of 0.01 rem or less in public areas outside the reactor building from vented fueled experiments. Fission gas radionuclides of Kr include 83m, 85m, 85, 87, 88, and 89. Fission gas radionuclides of Xe include 131m, 133m, 133, 135m, 135, 137, and 138. TEDE doses are calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19.
    e. For Specifications 3.8 d.iii and 3.8 d.iv, radiation doses were calculated as described in the Fueled Experiment Analysis Report for TS Amendment 19. Footnote (1) to TS 3.8.d.iv.3 specifies the required filter removal efficiency. Meeting Specification 3.8.d.iii and 3.8 d.iv gives a TEDE less than 1 rem and the Total Organ Dose-Equivalent to the thyroid (TODE) less than 25 rem to occupants inside the reactor building.
    f. Specification 3.8.e requires all specifications pertaining to criticality control given in TS 5.3 be satisfied thus ensuring that fueled experiments are stored in sub-critical configurations.

g. 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.

3.9 Primary Coolant

Applicability

This specification applies to the water quality and flow path of the primary coolant.

Objective

The objective is to ensure that primary coolant quality be maintained to acceptable values in order to reduce the potential for corrosion and limit the buildup of activated contaminants in the primary piping and pool.

Specification

The reactor shall not be operated unless the pool water meets the following limits:

a. The resistivity shall be ≥500 kΩ·cm.

b. The pH shall be within the range of 5.5 to 7.5.

Bases

The limits on resistivity are based on reducing the potential for corrosion in the primary piping or pool liner and to reduce the potential for activated contaminants in these systems.

4.0 SURVEILLANCE REQUIREMENTS

All surveillance tests required by these specifications are scheduled as described; however, some system tests may be postponed at the required intervals if that system or a closely associated system is undergoing maintenance. Any pending surveillance tests will be completed prior to reactor startup. Any surveillance item(s) which require reactor operation will be completed immediately after reactor startup. Surveillance requirements scheduled to occur during extended operation which cannot be performed while the reactor is operating may be deferred until the next planned reactor shutdown.

The intent of the surveillance interval (e.g., annually, but not to exceed fifteen months) is to maintain an average cycle, with occasional extensions as allowed by the interval tolerance. If it is desired to permanently change the scheduled date of surveillance, the particular surveillance item will be performed at an earlier date and the associated interval normalized to this revised earlier date. In no cases will permanent scheduling changes, which yield slippage of the surveillance interval routine scheduled date, be made by using the allowed interval tolerance.

4.1 Fuel

Applicability

This specification applies to the surveillance requirement for the reactor fuel.

Objective

The objective is to monitor the physical condition of the PULSTAR fuel.

Specification

a. All fuel assemblies shall be visually inspected for physical damage biennially but at intervals not to exceed thirty (30) months.

b. The reactor will be operated at such power levels necessary to determine if an assembly has had fuel pin cladding failure.

Bases

Each fuel assembly is visually inspected for physical damage that would include corrosion of the end fitting, end box, zircaloy box, missing fasteners, dents, severe surface scratches, and blocked coolant channels.

Based on a long history of prototype PULSTAR operation in conjunction with primary coolant analysis, biennial inspections of PULSTAR fuel to ensure fuel assembly integrity have been shown to be adequate for Zircoloy-2 (Zr-2) clad fuel. Any assembly that appears to have leaking fuel pin(s) will be disassembled to

confirm and isolate damaged fuel pins. Damaged fuel pins will be logged as such and permanently removed from service.

4.2 Control Rods

Applicability

This specification applies to the surveillance requirements for the control rods, shim rod, and control rod drive mechanisms (CRDM).

Objective

The objective is to assure the operability of the control rods and shim rod, and to provide current reactivity data for use in verifying adequate shutdown margin.

Specification

a. The reactivity worth of the shim rod and each control rod shall be determined annually but at intervals not to exceed fifteen (15) months for the steady state core in current use. The reactivity worth of all rods shall be determined for any new core or rod configuration, prior to routine operation.

b. Control rod drop times[1] and control rod drive times shall be determined:

   i. Annually but at intervals not to exceed fifteen (15) months.

   ii. After a control assemblyis moved to a new position in the core or after maintenance or modification is performed on the control rod drive mechanism.

c. The control rods shall be visually inspected biennially but at intervals notto exceed thirty (30) months.

d. The values of excess reactivity and shutdown margin shall be determined monthly, but at intervals not to exceed six (6) weeks, and for new core configurations.

[1] Applies only to magnetically coupled rods.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worths of experiments inserted in the core. The measurement of reactivity worths on an annual basis provides a correction for the slight variations expected due to burnup. This frequency of measurement has been found acceptable at similar research reactor facilities, particularly the prototype PULSTAR which has a similar slow change of rod value with burn-up.

Control rod drive and drop time measurements are made to determine whether the rods are functionally operable. These time measurements may also be utilized in reactor transient analysis.

Visual inspections include: detection of wear or corrosion in the rod drive mechanism; identification of deterioration, corrosion, flaking or bowing of the neutron absorber material; and verification of rod travel setpoints.

Control rod surveillance procedures will document proper control rod system reassembly after maintenance and recorded post-maintenance data will identify significant trends in rod performance.

4.3 Reactor Instrumentation and Safety Systems

Applicability

This specification applies to the surveillance requirements for the Reactor Safety System and other required reactor instruments.

Objective

The objective is to assure that the required instrumentation and Safety Systems will remain operable and will prevent the Safety Limits from being exceeded.

Specification

a. A channel check of each measuring channel in the RSS shall be performed daily when the reactor is in operation.

b. A channel test of each channel in the RSS shall be performed prior to operation each day, or prior to each operation extending more than one day.

c. A channel calibration of the N-16 Channel shall be made semi-annually, but at intervals not to exceed seven and one-half (7½) months. A calorimetric measurement shall be performed to determine the N-16 detector current associated with full power operation.

d. A channel calibration of the following channels shall be made semi-annually but at intervals not to exceed seven and one-half (7½) months.[(1)]

   i. Pool Water Temperature
   ii. Primary Cooling and Flow Monitoring (Flapper)
   iii. Pool Water Level
   iv. Primary Heat Exchanger Inlet and Outlet Temperature
   v. Safety and Linear Power Channels

[(1)] A channel calibration shall also be required after repair of a channel component that has the potential of affecting the calibration of the channel.

Bases

The daily channel tests and checks will assure the Reactor Safety Systems are operable and will assure operations within the limits of the operating license. The semi-annual calibrations will assure that long term drift of the channels is corrected. The calorimetric calibration of the reactor power level, in conjunction with the N-l6 Channel, provides a continual reference for adjustment of the Linear, Log N and Safety Channel detector positions.

4.4 Radiation Monitoring Equipment

Applicability

This specification applies to the surveillance requirements for the area and stack effluent radiation monitoring equipment.

Objective

The objective is to assure that the radiation monitoring equipment is operable.

Specification

a. The area and stack monitoring systems shall be calibrated annually but at intervals not to exceed fifteen (15) months.

b. The setpoints shall be verified monthly, but shall not exceed (6) weeks.

c. Vented fueled experiment exhaust gas radiation and flow monitors shall be calibrated prior to initial operation of the experiment and annually, not to exceed 15 months, thereafter for as long as the experiment is in operation.

d. Filter replacements for vented fueled experiments shall be biennial, but shall not exceed 30 months, and shall have a removal efficiency for iodine adsorption of 0.95 or greater at the specified flow rates used.

e. Leak testing of the filter housing(s) used for vented fueled experiments is performed after new filters are installed and prior to initial use; e.g. by pressure testing.

Bases

These systems provide continuous radiation monitoring of the Reactor Building with a check of readings performed prior to and during reactor operations.

The monthly verification of the setpoints in conjunction with the annual calibration is adequate to identify long term variations in the system operating characteristics.

Filter replacement of 2 years to 2.5 years (30 months), is consistent with TS 4.5 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.

4.5 Confinement and Main HVAC System

Applicability

This specification applies to the surveillance requirements for the confinement and main HVAC systems.

Objective

The objective is to assure that the confinement system is operable.

Specification

a. The confinement and evacuation system shall be verified to be operable within seven (7) days prior to reactor operation.

b. Operability of the confinement system on auxiliary power will be checked monthly but at intervals not to exceed six (6) weeks.(1)

c. A visual inspection of the door seals and closures, dampers and gaskets of the confinement and ventilation systems shall be performed semi-annually but at intervals not to exceed seven and one-half (7½) months to verify they are operable.

d. The control room differential pressure (dp) gauges shall be calibrated annually but at intervals not to exceed fifteen (15) months.

e. The confinement filter train shall be tested biennially but at intervals not to exceed thirty (30) months and prior to reactor operation following confinement HEPA or carbon adsorber replacement. This testing shall include iodine adsorption, particulate removal efficiency and leak testing of the filter housing.(2)

f. The air flow rate in the confinement stack exhaust duct shall be determined annually but at intervals not to exceed fifteen (15) months. The air flow shall be not less than 600 CFM.

(1) Operation must be verified following modifications or repairs involving load changes to the auxiliary power source.

(2) Testing shall also be required following major maintenance of the filters or housing.

Bases

Surveillance of this equipment will verify that the confinement of the Reactor Building is maintained as described in Section 5 of the SAR.

4.6 Primary and Secondary Coolant

Applicability

This specification applies to the surveillance requirement for monitoring the radioactivity in the primary and secondary coolant.

Objective

The objective is to monitor the radioactivity in the pool water to verify the integrity of the fuel cladding and other reactor structural components. The secondary water analysis is used to confirm the boundary integrity of the primary heat exchanger.

Specification

a. The primary coolant shall be analyzed bi-weekly, but at intervals not to exceed eighteen (18) days. The analysis shall include gross beta/gamma counting of the dried residue of a one (1) liter sample or gamma spectroscopy of a liquid sample, neutron activation analysis (NAA) of an aliquot, and pH and resistivity measurements.

b. The secondary coolant shall be analyzed bi-weekly, but at intervals not to exceed eighteen (18) days. This analysis shall include gross beta/gamma counting of the dried residue of a one (1) liter sample or gamma spectroscopy of a liquid sample.

Bases

Radionuclide analysis of the pool water samples will allow detection of fuel clad failure, while neutron activation analysis will give corrosion data associated with primary system components in contact with the coolant. Refer to SAR Section 10. The detection of activation or fission products in the secondary coolant provides evidence of a primary heat exchanger leak. Refer to SAR Section 10.

5.0. DESIGN FEATURES

5.1. Reactor Fuel

a. The reactor fuel shall be UO_2 with a nominal enrichment of 4% or 6% in U-235, zircaloy clad, with fabrication details as described in the Safety Analysis Report.

b. Total burn-up on the reactor fuel is limited to 20,000 MWD/MTU.

5.2. Reactor Building

a. The reactor shall be housed in the Reactor Building, designed for confinement. The minimum free volume in the Reactor Building shall be 2.25×10^9 cm^3 (refer to SAR Section 13 analysis).

b. The Reactor Building ventilation and confinement systems shall be separate from the Burlington Engineering Laboratories building systems and shall be designed to exhaust air or other gases from the building through a stack with discharge at a minimum of 100 feet above ground level.

c. The openings into the Reactor Building are the truck entrance door, personnel entrance doors, and air supply and exhaust ducts.

d. The Reactor Building is located within the Burlington Engineering Laboratory complex on the north campus of North Carolina State University at Raleigh, North Carolina. Restricted Areas as defined in 10 CFR Part 20 include the Reactor Bay, Ventilation Room, Mechanical Equipment Room, Primary Piping Vault, and Waste Tank Vault. The PULSTAR Control Room is part of the Reactor Building, however it is also a controlled access area and a Controlled Area as defined in 10 CFR Part 20. The facility license applies to the Reactor Building and Waste Tank Vault. Figure 5.2-1 depicts the licensed area as being within the operations boundary.

5.3. Fuel Storage

Fuel, including fueled experiments and fuel devices not in the reactor, shall be stored in a geometrical configuration where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water except in cases where a fuel shipping container is used, then the licensed limit for the k_{eff} limit of the container shall apply.

5.4 Reactivity Control

Reactivity control is provided by four neutron absorbing blades. Each control blade is nominally comprised of 80 percent silver, 15 percent indium, and 5 percent cadmium with nickel cladding. Three of these neutron absorbing blades are magnetically coupled and have scramming capability. The remaining neutron absorbing blade is non-scrammable. One of the scrammable rods may be used for automatic servo-control of reactor power. When in use, the servo-control maintains a constant power level as indicated by the Linear Power Channel.

5.5 Primary Coolant System

The primary coolant system consists of the aluminum lined reactor tank, a N-16 delay tank, a pump, and heat exchanger, and associated stainless steel piping. The nominal capacity of the primary system is 15,600 gallons. Valves are located adjacent to the biological shield to allow isolation of the pool, and at major components in the primary system to permit isolation.

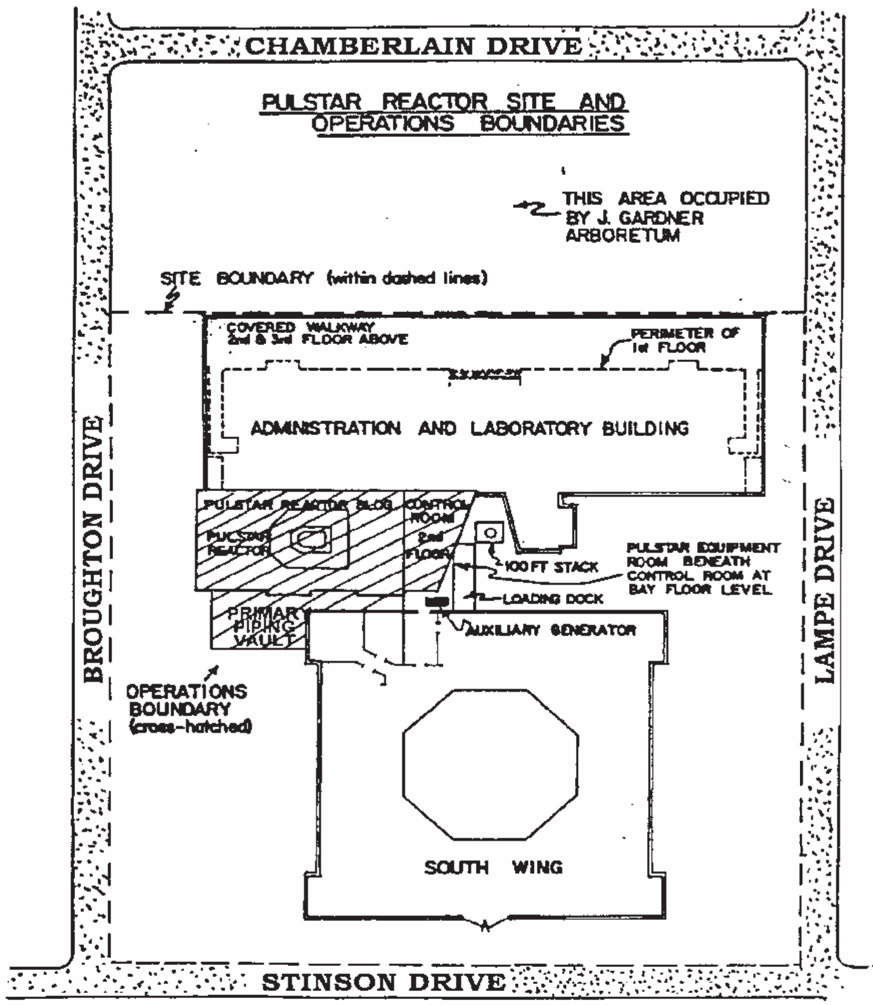


Figure 5.2- 1: NCSU PULSTAR Reactor Site Map

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

The reactor facility shall be an integral part of the Department of Nuclear Engineering of the College of Engineering of North Carolina State University. The reactor shall be related to the University structure as shown in Figure 6.1-1.

6.1.1 Organizational Structure:

The reporting chain is given in Figure 6.1-1. The following specific organizational levels (as defined by ANSI/ANS-15.1-1990) and positions shall exist at the PULSTAR Facility:

Level 1 – Administration

This level shall include the Chancellor, the Dean of the College of Engineering, and the Nuclear Engineering Department Head. Within three months of appointment, the Nuclear Engineering Department Head shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the facility.

Level 2 – Facility Management

This level shall include the Nuclear Reactor Program (NRP) Director. The NRP Director is responsible for the safe and efficient operation of the facility as specified in the facility license and Technical Specifications, general conduct of reactor performance and NRP operations, long range development of the NRP, and NRP personnel matters. The NRP Director evaluates new service and research applications, develops new facilities and support for needed capital investments, and controls NRP budgets. The NRP Director works through the Manager of Engineering and Operations to monitor daily operations and with the Reactor Health Physicist to monitor radiation safety practices and regulatory compliance. The minimum qualifications for the NRP Director are a Master of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. Within three months of appointment, the NRP Director shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the facility. The NRP Director is a faculty member and reports to the Nuclear Engineering Department Head.

Level 3 – Manager of Engineering and Operations

The Manager of Engineering and Operations (MEO) performs duties as assigned by the NRP Director associated with the safe and efficient operation of the facility as specified in the facility license and Technical Specifications. The MEO is responsible for coordination of operations, experiments, and maintenance at the facility, including reviews and approvals of experiments as defined in Technical Specification 1.2.9 and 6.5, and making minor changes to procedures as stated in Technical Specification 6.4. The MEO shall receive appropriate facility specific training within three months of appointment and be certified as a Senior Reactor Operator within one year of appointment. The minimum qualifications for the MEO are a Bachelor of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. The MEO reports to the NRP Director.

Level 4 – Operating and Support Staff

This level includes licensed Senior Reactor Operators (SRO), licensed Reactor Operators (RO), and other personnel assigned to perform maintenance and technical support of the facility. Senior Reactor Operators and Reactor Operators are responsible for assuring that operations are conducted in a safe manner and within the limits prescribed by the facility license and Technical Specifications, applicable Nuclear Regulatory Commission regulations, and the provisions of the Radiation Safety Committee and Reactor Safety and Audit Committee. All Senior Reactor Operators shall have three years of nuclear experience and shall have a high school diploma or successfully completed a General Education Development test. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of nuclear experience as applicable to research reactors for Senior Reactor Operators. Other Level 4 personnel shall have a high school diploma or shall have successfully completed a General Education Development test. All Level 4 personnel report to the Manager of Engineering and Operations.

Reactor Health Physicist

The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP shall have a high school diploma or shall have successfully completed a General Education Development test and have three years of relevant experience in applied radiation safety. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of experience in radiation safety as

applicable to research reactors. The RHP reports directly to the Nuclear Engineering Department Head and is independent of the campus Radiation Safety Division as shown in Figure 6.1-1.

6.1.2 Responsibility

Responsibility for the safe operation of the PULSTAR Reactor shall be with the chain of command established in Figure 6.1-1.

Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, the Technical Specifications, and federal regulations.

In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon the appropriate qualifications.

6.1.3 Minimum Staffing

The minimum staffing when the reactor is not secured shall be:

a. A licensed reactor operator or senior reactor operator shall be present in the Control Room.

b. A Reactor Operator Assistant (ROA), capable of being at the reactor facility within five (5) minutes upon request of the reactor operator on duty.

c. A Designed Senior Reactor Operator (DSRO). This individual shall be readily available on call, meaning:

    i. Has been specifically designated and the designation known to the reactor operator on duty.

    ii. Keeps the reactor operator on duty informed of where he may be rapidly contacted and the telephone number.

    iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15 mile radius).

d. A Reactor Health Physicist or his designated alternate. This individual shall also be on call, under the same limitations as prescribed for the Designed Senior Reactor Operator under Specification 6.1.3.c.

6.1.4 Senior Reactor Operator Duties

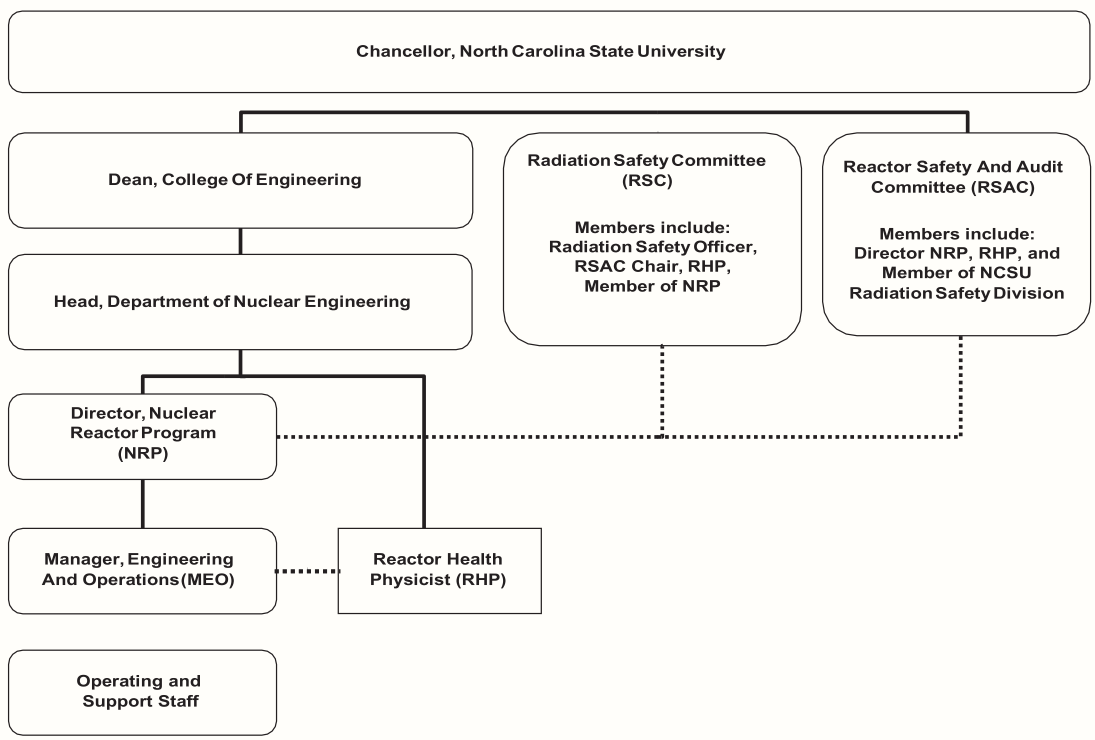
The following events shall require the presence of a licensed Senior Reactor Operator at the facility or its administrative offices:

a. Initial startup and approach to power.
b. All fuel or control rod relocations within the reactor core or pool.
c. Relocation of any in-core experiment with a reactivity worth greater than one dollar (730 pcm).
d. Recovery from unplanned or unscheduled shutdown or significant power reduction (documented verbal concurrence from a licensed Senior Reactor Operator is required).

6.1.5 Selection and Training

All operators will undergo a selection, training and licensing program prior to unsupervised operation of the PULSTAR reactor. All licensed operators will participate in a requalification program, which will be conducted over a period not to exceed two (2) years. The requalification program will be followed by successive two (2) year programs.

Figure 6.1-1: NCSU PULSTAR Reactor Organizational Chart



NOTES: Line of direct communication ———

Line of advice and liaison – – –

Nuclear Reactor Program (NRP) includes:

- Director, NRP
- Manager, Engineering and Operations
- Operating and Support Staff

Reactor Health Physicist (RHP) reports to the Head, Department of Nuclear Engineering and serves both the NRP and Department of Nuclear Engineering.

Communication on reactor operations, experiments, radiation safety, and regulatory compliance occurs between the NRP, RHP, Reactor Safety and Audit Committee, Radiation Safety Committee, and campus Radiation Safety Division as described in these Technical Specifications and facility procedures.

6.2 Review and Audit

The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices, including the nuclear reactor, at the University are in compliance with state and federal licenses and all applicable regulations. The RSC reviews and approves all experiments involving the potential release of radioactive material conducted at the University and provides oversight of the University Radiation Protection Program. The RSC is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by RSAC and the Nuclear Reactor Program (NRP).

RSAC has the primary responsibility to ensure that the reactor is operated and used in compliance with the facility license, Technical Specifications, and all applicable regulations. RSAC performs an annual audit of the operations and performance of the NRP.

6.2.1 RSC and RSAC Composition and Qualifications

a. RSC shall consist of members from the general faculty who are actively engaged in teaching or research involving radioactive materials or radiation devices. RSC may also include non-faculty members who are knowledgeable in nuclear science or radiation safety. RSC membership shall include the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

b. RSAC shall consist of at least five individuals who have expertise in one or more of the component areas of nuclear reactor safety. These include Nuclear Engineering, Nuclear Physics, Health Physics, Electrical Engineering, Chemical Engineering, Material Engineering, Mechanical Engineering, Radiochemistry, and Nuclear Regulatory Affairs.

   At least three of the RSAC members are appointed from the general faculty. The faculty members shall be as follows:

   i. NRP Director

   ii. One member from an appropriate discipline within the College of Engineering

   iii. One member from the general faculty

The remaining RSAC members are as follows:

iv. Reactor Health Physicist (RHP)

v. Member from the campus Radiation Safety Division of the Environmental Health and Safety Center

vi. One additional member from an outside nuclear related establishment may be appointed

vii. At the discretion of RSAC, specialist(s) from other universities and outside establishments may be invited to assist in its appraisals.

The NRP Director, RHP, and a member from the campus Radiation Safety Division of the Environmental Health and Safety Center are permanent members of RSAC.

6.2.2 RSC and RSAC Rules

a. RSC and RSAC committee member appointments are made by University Management for terms of three (3) years.

b. RSC shall meet as required by the broad scope radioactive materials license issued to the University by the State of North Carolina. RSC may also meet upon call of the committee Chair.

c. RSAC shall each meet at least four (4) times per year, with intervals between meetings not to exceed six months. RSAC may also meet upon call of the committee Chair.

d. A quorum of RSC or RSAC shall consist of a majority of the full committee membership and shall include the committee Chair or a designated alternate for the committee Chair. Members from the line organization shown in Figure 6.1-1 shall not constitute a majority of the RSC or RSAC quorum.

6.2.3 RSC and RSAC Review and Approval Function

a. The following items shall be reviewed and approved by the RSC:

   i. All new experiments or classes of experiments that could result in the release of radioactivity.

   ii. Proposed changes to the facility license or Technical Specifications, excluding safeguards information.

b. The following items shall be reviewed and approved by the RSAC:

   i. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures which have safety significance meet facility license and Technical Specification requirements.

   ii. All new procedures and major revisions having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

   iii. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

   iv. Proposed changes to the facility license or Technical Specifications, including safeguards information.

c. The following items shall be reviewed by the RSC and RSAC:

   i. Violations of the facility license or Technical Specifications

   ii. Violations of internal procedures or instructions having safety significance.

   iii. Operating abnormalities having safety significance.

   iv. Reportable Events as defined in Specification 1.2.24.

Distribution of RSC summaries and meeting minutes shall include the RSAC Chair and Director of the Nuclear Reactor Program.

A summary of RSAC meeting minutes, reports, and audit recommendations approved by RSAC shall be submitted to the Dean of the College of Engineering, the Nuclear Engineering Department Head, the Director of the Nuclear Reactor Program, the RSC Chair, Director of Environmental Health and Safety, RSAC Chair, and the Manager of Engineering and Operations prior to the next scheduled RSAC meeting.

6.2.4 RSAC Audit Function

The audit function shall consist of selective, but comprehensive, examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations shall also be used as appropriate. The RSAC shall be responsible for this audit function. In no case shall an individual immediately responsible for the area perform an audit in that area. This audit shall include:

a. Facility operations for conformance to the facility license and Technical Specifications, annually, but at intervals not to exceed fifteen (15) months.

b. The retraining and requalification program for the operating staff, biennially, but at intervals not to exceed thirty (30) months.

c. The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, annually, but at intervals not to exceed fifteen (15) months.

d. The Emergency Plan and Emergency Procedures, biennially, but at intervals not to exceed thirty (30) months.

e. Radiation Protection annually, but at intervals not to exceed fifteen (15) months.

Deficiencies uncovered that affect reactor safety shall be immediately reported to the Nuclear Engineering Department Head, Director of the Nuclear Reactor Program, and the RSC.

The annual audit report made by the RSAC, including any recommendations, is provided to the RSC.

6.3 Radiation Safety

The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP reports directly to the Nuclear Engineering Department Head and is independent of the campus Radiation Safety Division as shown in Figure 6.1-1.

6.4 Operating Procedures

Written procedures shall be prepared, reviewed and approved prior to initiating any of the following:

a. Startup, operation and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the reactor.

c. Maintenance of major components of systems that could have an affect on reactor safety.

d. Surveillance checks, calibrations and inspections required by the facility license or Technical Specifications or those that may have an affect on the reactor safety.

e. Personnel radiation protection, consistent with applicable regulations and that include commitment and/or programs to maintain exposures and releases as low as reasonably achievable (ALARA).

f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

g. Implementation of the Emergency Plan and Security Plan.

Substantive changes to the above procedures shall be made effective only after documented review and approval by the RSAC and by the Manager of Engineering and Operations.

Minor modifications to the original procedures which do not change their original intent may be made by the Manager of Engineering and Operations, but the modifications shall be approved by the Director of the Nuclear Reactor Program within fourteen (14) days.

Temporary deviations from procedures may be made by Designed Senior Reactor Operator as defined by Specification 6.1.3.c or the Manager of Engineering and Operations, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to the Director of the Nuclear Reactor Program.

6.5 Review of Experiments

6.5.1 New (untried) Experiments

All new experiments or class of experiments, referred to as "untried" experiments, shall be reviewed and approved by the RSC, the RSAC, the Director of the Nuclear Reactor Program, Manager of Engineering and Operations, and the Reactor Health Physicist, prior to initiation of the experiment.

The review of new experiments shall be based on the limitations prescribed by the facility license and Technical Specifications and other Nuclear Regulatory Commission regulations, as applicable.

6.5.2 Tried Experiments

All proposed experiments are reviewed by the Manager of Engineering and Operations and the Reactor Health Physicist (or their designated alternates). Either of these individuals may deem that the proposed experiment is not adequately covered by the documentation and/or analysis associated with an existing approved experiment and therefore constitutes an untried experiment that will require the approval process detailed under Specification 6.5.1.

If the Manager of Engineering and Operations and the Reactor Health Physicist concur that the experiment is a tried experiment, then the request may be approved.

Substantive changes to previously approved experiments will require the approval process detailed under Specification 6.5.1.

6.6 Required Actions

6.6.1 Action to be Taken in Case of Safety Limit Violation

In the event a Safety Limit is violated:

a. The reactor shall be shutdown and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. The Safety Limit violation shall be promptly reported to the Director of the Nuclear Reactor Program, or his designated alternate.

c. The Safety Limit violation shall be reported to the Nuclear Regulatory Commission in accordance with Specification 6.7.1.

d. A Safety Limit violation report shall be prepared that describes the following:

   i. Circumstanced leading to the violation including, when known, the cause and contributing factors.
   ii. Effect of violation upon reactor facility components, systems, or structures and on the health and safety of facility personnel and the public.
   iii. Corrective action(s) to be taken to prevent recurrence.

The report shall be reviewed by the RSC and RSAC and any follow-up report shall be submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation.

6.6.2 Action to be Taken for Reportable Events (other than SL Violation)

In case of a Reportable Event (other than violation of a Safety Limit), as defined by Specification 1.2.24, the following actions shall be taken:

a. Reactor conditions shall be returned to normal or the reactor shall be shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operation shall not be resumed unless authorized by the Director of the Nuclear Reactor Program, or his designated alternate.

b. The occurrence shall be reported to the Director of the Nuclear Reactor Program, and to the Nuclear Regulatory Commission in accordance with Specification 6.7.1.

c. The occurrence shall be reviewed by the RSC and RSAC at their next scheduled meeting.

6.7 Reporting Requirements

6.7.1 Reportable Event

For Reportable Events as defined by Specification 1.2.24, there shall be a report not later than the following work day by telephone to the Nuclear Regulatory Commission Operations Center followed by a written report within fourteen (14) days that describes the circumstances of the event.

6.7.2 Permanent Changes in Facility Organization

Permanent changes in the facility organization involving either Level 1 or 2 personnel (refer to Specification 6.1.1) shall require a written report within thirty (30) days to the Nuclear Regulatory Commission Document Control Desk.

6.7.3 Changes Associated with the Safety Analysis Report

Significant changes in the transient or accident analysis as described in the Safety Analysis Report shall require a written report within thirty (30) days to the Nuclear Regulatory Commission Document Control Desk.

6.7.4 Annual Operating Report

An annual operating report for the previous calendar year is required to be submitted no later than March 31st of the present year to the Nuclear Regulatory Commission Document Control Desk. The annual report shall contain as a minimum, the following information:

a. A brief narrative summary:
   i. Operating experience including a summary of experiments performed.
   ii. Changes in performance characteristics related to reactor safety that occurred during the reporting period.
   iii. Results of surveillance, tests, and inspections.

b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality.

c. The number of emergency shutdowns and unscheduled SCRAMs, including reasons and corrective actions.

d. Discussion of the corrective and preventative maintenance performed during the period, including the effect, if any, on the safety of operation of the reactor.

e. A brief description, including a summary of the analyses and conclusions of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59.

f. A summary of the nature and amount of radioactive effluent released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge, including:

Liquid Waste (summarized by quarter)

i. Radioactivity released during the reporting period:
   1. Number of batch releases.
   2. Total radioactivity released (in microcuries).
   3. Total liquid volume required (in liters).
   4. Diluent volume required (in liters).
   5. Tritium activity released (in microcuries)
   6. Total (yearly) tritium released.
   7. Total (yearly) activity released.

ii. Identification of fission and activation products:

   Whenever the undiluted concentration of radioactivity in the waste tank at the time of release exceeds 2×10^{-5} µCi/ml, as determined by gross beta/gamma count of the dried residue of a one liter sample, a subsequent analysis shall also be performed prior to release for principle gamma emitting radionuclides. An estimate of the quantities present shall be reported for each of the identified nuclides.

iii. Disposition of liquid effluent not releasable to the sanitary sewer system:

   Any waste tank containing liquid effluent failing to meet the requirements of 10 CFR Part 20, Appendix B, to include the following data:
   1. Method of disposal.
   2. Total radioactivity in the tank (in microcuries) prior to disposal.
   3. Total volume of liquid in tank (in liters).

4. The dried residue of one liter sample shall be analyzed for the principle gamma-emitting radionuclides. The identified isotopic composition with estimated concentrations shall be reported. The tritium content shall be included.

Gaseous Waste

i. Radioactivity discharged during the reporting period (in curies) for:
   1. Gases
   2. Particulates, with half lives greater than eight days.
ii. The Airborne Effluent Concentration (AEC) used and the estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis. (AEC values are given in 10 CFR Part 20, Appendix B, Table 2.)

Solid Waste

i. The total amount of solid waste packaged (in cubic feet).
ii. The total activity involved (in curies).
iii. The dates of shipment and disposition (if shipped off-site).

g. A summary of radiation exposures received by facility personnel and visitors, including pertinent details of significant exposures.

h. A summary of the radiation and contamination surveys performed within the facility and significant results.

i. A description of environmental surveys performed outside the facility.

6.8 Retention of Records

Records and logs of the following items, as a minimum, shall be kept in a manner convenient for review and shall be retained as detailed below. In addition, any additional federal requirement in regards to record retention shall be met.

6.8.1 Records to be retained for a period of at least five (5) years

a. Normal plant operation and maintenance.

b. Principal maintenance activities.

c. Reportable Events.

d. Equipment and components surveillance activities as detailed in Specification 4.

e. Experiments performed with the reactor.

f. Changes to Operating Procedures.

g. Facility radiation and contamination surveys other than those used in support of personnel radiation monitoring.

h. Audit summaries.

i. RSC and RSAC meeting minutes.

6.8.2 Records to be retained for the life of the facility

a. Gaseous and liquid radioactive waste released to the environs.

b. Results of off-site environmental monitoring surveys.

c. Radiation exposures for monitored personnel and associated radiation and contamination surveys used in support of personnel radiation monitoring.

d. Fuel inventories and transfers.

e. Drawings of the reactor facility.

6.8.3 Records to be retained for at least one (1) license period of six (6) years

Records of retraining and requalification of certified operating personnel shall be maintained at all times the individual is employed, or until the certification is renewed.