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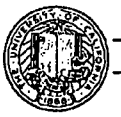

SANTA BARBARA • SANTA CRUZ

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August 1st, 2011

US Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555
Attention: Linh Tran, Senior Project Manager
Francis DiMeglio, Senior Project Manager

Re: **Docket 50-326 Relicense, RAI dated April 5, 2011 (TAC ME1579), additional response material - extension granted to August 1st, 2011.**

Dear Ms Tran, Mr DiMeglio:

A response was sent on June 24th to the referenced RAI. One remaining item requested was revision of the MHA discussion. A response is contained herein, along with other items discussed at our meeting in June.:

A. Attached is a complete revision of SAR 13.2 addressing the MHA for the UCI Reactor Facility and removing material from the 1968/1999 SAR that is no longer required.

The following changes to the input data were incorporated:

1. An increase in the total volume of the reactor room area used since the two lab areas are not separated in air volume, from 5×10^8 cm^3 to 6.6×10^8 cm^3.
2. An increase in the number of fuel elements across which the fission product inventory is distributed from 70 to 80 elements.
3. A decrease in the peak to average power density from 1.6 to 1.45 according to recent neutronic calculations submitted earlier.

These three revisions have the net effect of reducing the exposure risks associated with the MHA by 30%.

4. The newly revised ventilation system (described in an attachment of revised SAR section 3.6) increases the dilution of reactor effluent from an assumed factor of 40 used in the former SAR to a certain factor of almost 100. In addition this plume is even further removed from public (or other personnel) potential access. This reduces the potential exposure risk to off-site persons.

5. A new section 13.2.4.2 addresses the potential for exposure to personnel in areas adjacent to the reactor during the MHA. These results are much lower, in

A020
NRR

disagreement with the assessment submitted in the facility original SAR, but the error made then does not seem to be reproduced in this analysis.

Our lower dose risk assessments are submitted herewith. However because these are recent conclusions, based on simplified models, we have requested external review of these conclusions by General Atomic. The results are not yet available, but when their assessments are available, we will immediately submit them to NRC, however we believe the present conclusions will be supported, and perhaps even lowered by their, more detailed, analysis.

B. A revised section 6.3.2.2 addressing the argon-41 production and release at the facility is attached. This is in support of the proposed technical specification 3.7.2. setting the Argon-41 release concentration limit when released to the building exhaust to 1×10^{-6} microcuries/mL.

C. Two revisions to the proposed Technical Specifications for the facility that was last reviewed in June are attached. The complete Technical Specifications have now received approval from the UCI Reactor Operations Committee.
(1) revision to the argon concentration limit in TS. 3.7.2, as mentioned.
(2) revision to the surveillance for the seismic scram switch in TS 4.

I declare under penalty of perjury that the foregoing and the attached are true and correct to my knowledge.

Executed on August 1st, 2011

G. E. Miller

Dr. George E. Miller

University of California – Irvine (UCAI
License No. R – 116
Docket No. 05000326

Response to RAIs Dated April 5, 2011 (TAC ME1579), Additional Response Material – Extension Granted to August 1st, 2011

Redacted Version*

Security-Related Information Removed

* Redacted text and figures blacked out or denoted by brackets.

6.3.2.2 Argon-41

In the UCINRF only the pneumatic tube system, the central thimble, and the rotary specimen rack contain air. There are no permanent beam tubes or exposure rooms. ^{41}Ar is produced by neutron capture by ^{40}Ar in the air and is the most significant hazard during normal operation. As a noble gas with a short half-life, argon-41 contributes to internal and external exposure from cloud submersion, but not to any committed dose.

Experience with this and other TRIGA reactors has shown that no detectable release of ^{41}Ar occurs from the rotary specimen rack or the central thimble during full power operation because there is no significant "flushing" of these regions. A small amount is released from the pool water.

The pneumatic tube (PT) systems discharge directly into the main reactor room exhaust duct. The concentration in the diluted discharge of the main PT system has been measured to be 6×10^{-8} microcuries/mL with the PT blower running and the reactor operating at 250 kilowatts. The flow rate of this discharge is 1×10^{8} mL per minute. Measurements were also made of ^{41}Ar over the pool water after the reactor had been operated for several hours. Based on dilution estimates this is found to add less than 1×10^{-8} microcurie per mL to the exhaust discharge. Thus the estimated maximum release rate from the facility exhaust during pneumatic transfer system operation is 7×10^{-8} microcuries per mL, and with reactor operation without the PT in use is 1×10^{-8} microcuries/mL.

For comparison, the 10CFR 20 Appendix B values for ^{41}Ar are:

Table 1, Col. 3: DAC (inhalation) = 3×10^{-6} microcuries/mL

Table 2, Col. 1: Air (effluent) = 1×10^{-8} microcuries/mL

In terms of occupational exposure, the individual in the reactor room could be in a local concentration at the DAC directly over the pool during prolonged reactor operation. In the main area, the concentration is at 1×10^{-8} μc/mL only, so TEDE from this source is acceptable.

6.3.2.2.1 Argon-41 Dispersion in the Atmosphere

It was calculated above that the maximum release of ^{41}Ar into the building exhaust flow is at a concentration of 7×10^{-8} μc/mL. It is shown in SAR section 3.6.1 that the building ventilation system provides a dilution by at least a factor of 100 when operational. Furthermore the ventilation design extends the plume height to an effective 120 feet above roof level compared to the 24 foot height of the actual mechanical fan system. Thus release is achieved at a height above ground level of well over 200 feet. Thus the calculated maximum plume concentration is approximately 7×10^{-10} μc/mL, well below the 10CRF20 Appendix B limit of 1×10^{-8} μc/mL. This is only when the pneumatic transfer system is operating. Continuous operation of the reactor itself results in a concentration of only 1×10^{-10} μc/mL.

To estimate the hazard to the community of such a constant source of gaseous activity requires the assumption of a Gaussian form for the plume dispersion[1], certain values for the diffusion parameters and a mean wind velocity corresponding to a certain meteorological condition. This method tends to give a conservative account since it is improbable that such parameters will remain constant over long periods of time. Any variation will tend to reduce the dose received at any particular location away from the source.

In order to obviate some uncertainty, an attempt is made to establish a lower limit of exposure by placing a $CaSO_4$/Dy environmental dosimeter pack at the effluent stack to represent the exposure that an individual at that location would receive. The pack is exchanged and read quarterly. Within the limits of error, the readings over many years have been consistently within the expected background variation (approximately 10 mrem computed annually). This meets the requirement of 20.1101 (d) without invoking dilution from mixing beyond the exhaust plume.

For reporting purposes, computations are made each year based on the conservative assumption that all reactor runs are at full power. The number of minutes of PT operation is recorded, and the full annual production of ^{41}Ar computed from the measurements of local concentrations listed above. The upgrade of the ventilation system carried out in December 2010 has provided even greater assurance of dilution before release to the environment. Prior to this, assumptions were made that dilution of a factor of 40 was likely. Now, the dilution of 100 is assured, and the release is at a greater height of 200 feet (almost 10 meters) above ground level.

The maximum release value is attained only during operation of the pneumatic transfer system, typically for less than 500 minutes per year. Regular steady state operation generates less than 1/7th of that amount.

6.3.2.2.2 Concentration of ^{41}Ar at the Air Intakes of Rowland Hall.

The safety of persons inside the building must also be considered. The possibility of contamination of the air taken into the main building by the reactor effluent is considered here.

The new ventilation system of Rowland Hall was specifically designed to carry and dilute fumes from fume hood vents from chemistry research and teaching laboratories, and avoid any backflow or recycle into the building air in order to meet OSHA standards. Because of the intense odor of certain chemicals, failure of the system is readily detectable! The system is continuously monitored at the UCI Central Plant and alarms provide alerts in the event of failure or reduced performance. The provision of flow rate and pressure differential information in the reactor control room enable establishment of ventilation performance as limiting conditions for operation of the reactor so that operations will be discontinued in the event of any ventilation system malfunction.

These changes and policies help to assure that air from the reactor ventilation is diluted and kept away from this and other local buildings.

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[1] F. A. Gifford, Nuclear Safety, V. 2, No. 2, Dec. 1960, p. 56, and Nuclear Safety, Vol. 2, No, 4, July 1961, p. 47.

3.6 VENTILATION SYSTEM

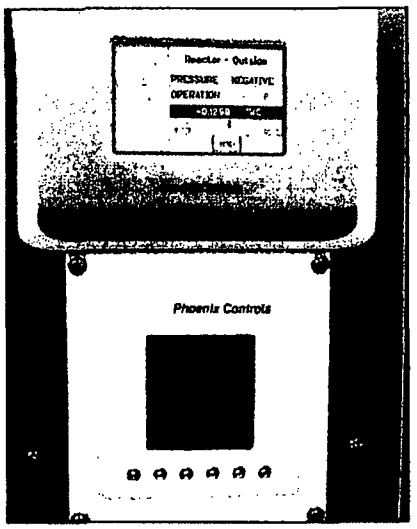
3.6.1 Normal Operation

The ventilation system for this area is diagrammed in Fig. 3-1. In normal operation, fresh air is drawn into the building at the service floor level via two air handling units equipped with filters and hot and chilled water coils. The airflow is split to form a hot and cold deck supply distribution system for the building. Three supply valves supply the control room, the reactor room and the rear laboratories respectively. A local thermostat controls mixing dampers at the entrance to each supply area to maintain room temperature. Temperatures and static pressures in the main supply ducts are maintained by automatic controls on the Phoenix valve flow dampers. Exhaust ducts are controlled in a similar manner to maintain the negative pressures requirements. The complete system is under the control of the Central Plant which is manned 24 hours a day.

In the reactor area system, in addition to the mixing control dampers, static pressure controller regulates the supply air to the reactor room so that the negative pressure requirement is met; an additional controller operates in emergency mode as described below.

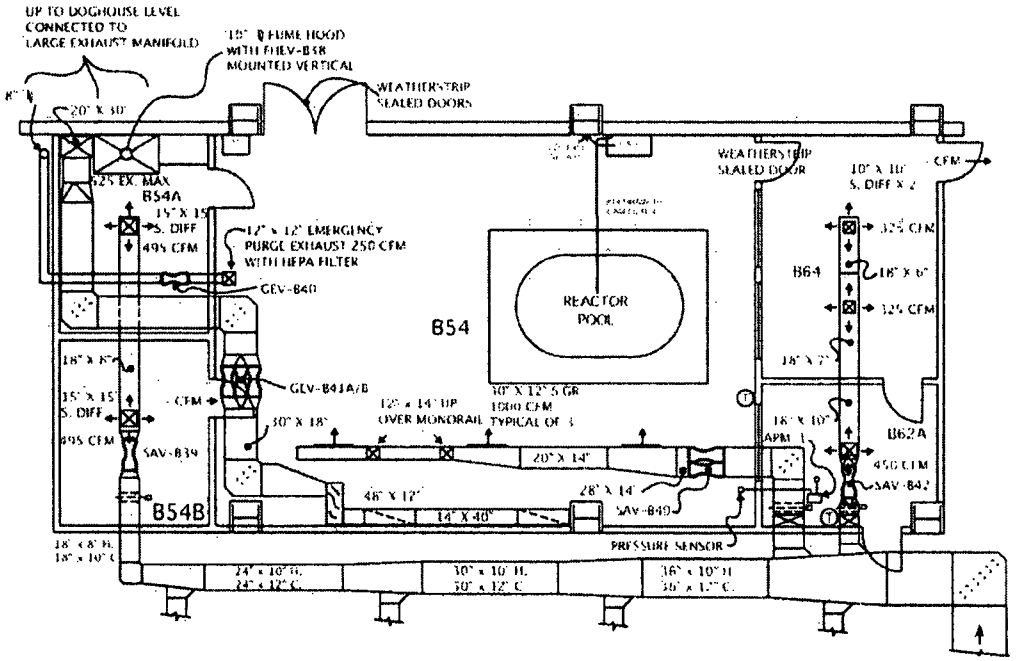
The system of supply and exhaust controllers operates 24/7 but provision is made for lowering exhaust and supply flow rates during periods when the facility is unoccupied. This change-over is accomplished manually using an intelligent control box in the control room. The negative pressure differential is maintained in both occupied and unoccupied modes.

When the facility is occupied, and/or the reactor is in operation, the main reactor area exhaust is designed to provide a minimum total exhaust flow rate of 3500 cfm , with the fume hood adding over 500 cfm, for a total of over 4,000 cfm. Dilution of the exhaust from the facility and other building laboratories is provided at the roof level by the three building exhaust fans diagrammed in Fig. 3-1 and shown in detail in Fig. 3-2. The exhaust fans installed are 66 inch diameter high plume fan model Axijet-S 6600 which provide exhaust flow rates of 80,000 cfm. The fan design provides an upward induced inflow of an additional 48,000 cfm of roof air for a combined total of 128,000 cfm. This extends the plume height to an effective 120 feet above roof level compared to the 24 foot height of the mechanical system. In full normal operation, a dilution factor of 3 x 128,000/ 4,000 or a factor of 96 is achieved at a height above ground level of over 200 feet. If any discharge, such as that from the pneumatic transfer system blower, is only into the main exhaust, the dilution is by a factor of 110. Because the building houses many chemical laboratories, the system design is provided to assure little likelihood of personnel exposure due to feedback of chemical fumes, including radioactive materials, either directly, or into building air intakes which are at building ground level.

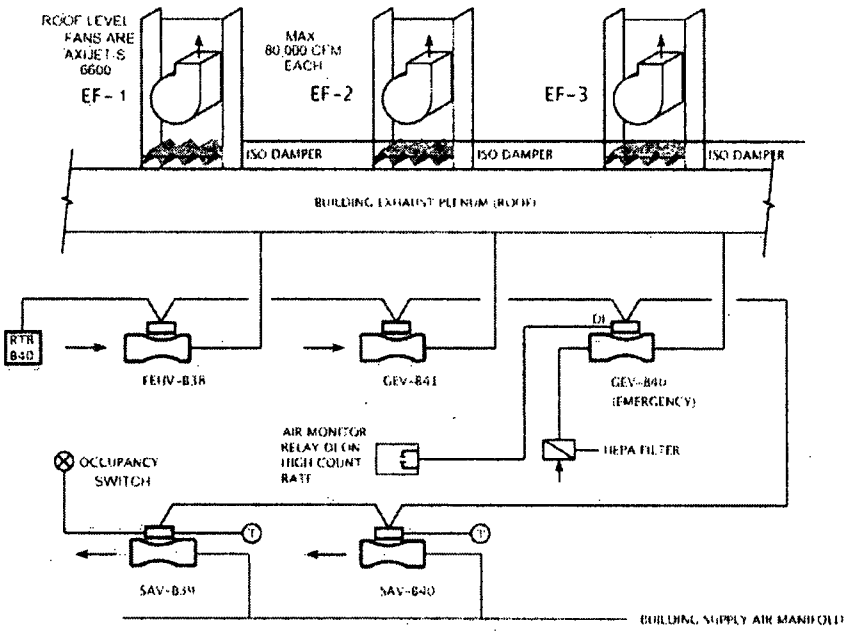


The flow rates and room pressure differential values are displayed locally and at the Central Plant.

The local display is illustrated here. The upper unit shows the pressure differentials, the lower unit displays flow rates, temperatures, and other system information.

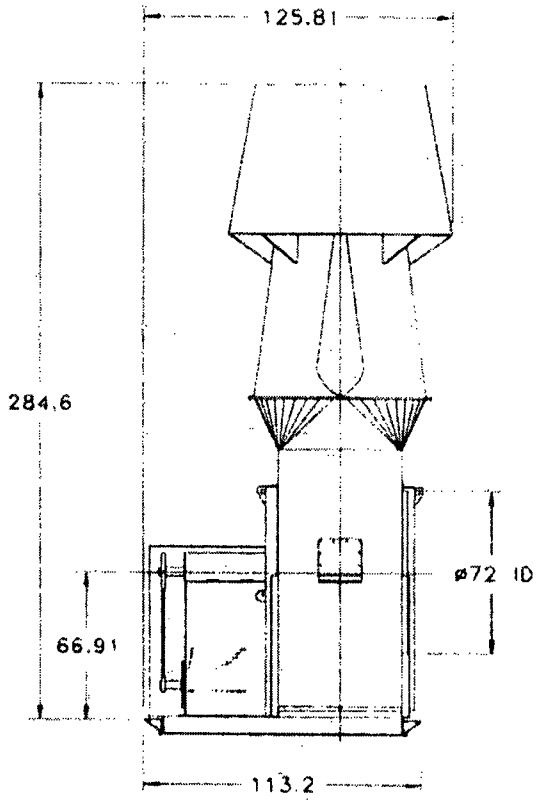


NEW AIR CONDITIONING AND VENTILATION PLAN
(DEC. 2010)



NEW VENTILATION SYSTEM CONTROL SCHEMATIC
(DEC. 2010)

Figure 3-1 Reactor facility ventilation and air conditioning system



Sys. No.	EF-1, EF-2, EF-3
Drawing	14336-1 A
Revision	0

Model	AXIJET
Fan Size	6600
Dia.[in]	66.00
CFM	80000
SP	4
BHP	108.78
RPM	694

	10	[mph]
EH =	120.1	[feet]
PH =	96.6	[feet]
NV =	7273	[fpm]
TF =	128000	[cfm]
TS =	11991	[fpm]
T =	70	[°F]
ALT =	0	[feet]

CLASS II WHEEL

Air performance

TF=Total Flow, NV=Nozzle Velocity, PH=Plume Height
EH=Effective Height(EH=PH+Fan Height), TS=Tip Speed

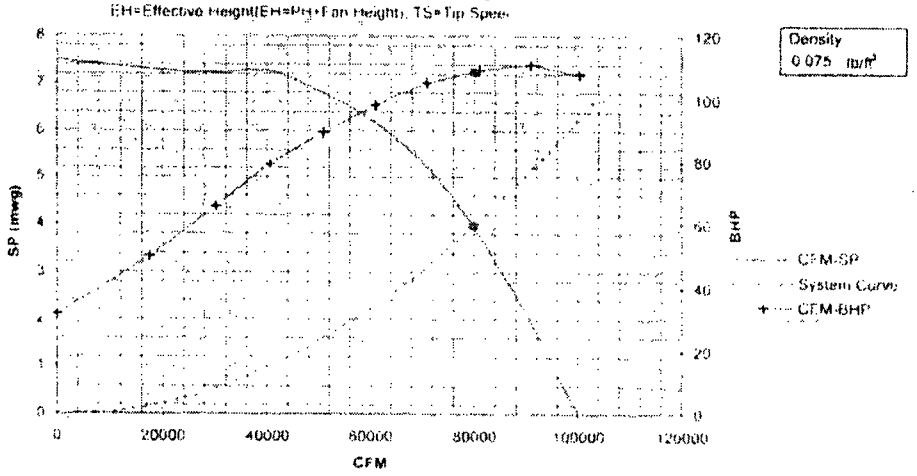


Figure 3-2 Rowland Hall roof exhaust fan Axijet S 6600 details

3.6.2 Emergency operation

In the unlikely event of an accidental release of particulate or gaseous radioactivity into the reactor room air it is desirable that this should be exhausted slowly to the atmosphere to allow dilution, and filtered to remove any radioactive particulates. Thus an emergency purge exhaust is installed, actuated by a signal from a continuous air particulate monitor (CAM described below (3.7.1)). Emergency mode can also be actuated via a push button on the control console.

In emergency mode, air is exhausted from the reactor room at a rate of about 250 cfm through a Magnamedia Beryllium-collector style filter. At the same time, supply valves (Fig. 3-1) to the reactor room and laboratories are closed and the regular reactor room exhaust and the fume hood exhaust control valves are closed. The pneumatic transfer system blower is disabled, and audio and visual alarms indicate that an emergency condition has been actuated. Flow rates can be monitored at the control room ventilation system controller to assure this has been accomplished.

In the emergency condition, air inflow is only the natural leakage rate of the area. This should be less than the 250 cfm exhaust capacity so that a small negative pressure differential is maintained. This can be validated at the system display in the control room.

3.7.2 Effluents

Applicability. This specification applies to the release rate of ^{41}Ar.

Objective. To assure that concentration of ^{41}Ar in accessible unrestricted areas shall be below the applicable limits of 10 CFR Part 20.

Specification. The annual average concentration of ^{41}Ar released into the building exhaust shall not exceed 1×10^{-6} μCi/mL.

Basis. It is shown in sections 3.6.1 and 6.3.2.2 of the SAR that dilution of at least a factor of 100 is provided by the building exhaust fan system during normal reactor operation and further that the discharge plume is at an inaccessible location and would experience additional dilution before any exposure to an individual would be experienced . This will assure that the requirement of 10 CFR 20 Appendix B, Table 2 for release to unrestricted areas of a concentration of ^{41}Ar no greater than 1×10^{-8} μCi/mL is met.

4.2 Reactor Control and Safety Systems

Applicability. This specification applies to the surveillance requirements for the reactor control and safety systems.

Objective. The objective is to verify performance and operability of those systems and components which are directly related to reactor safety.

Specifications.

a. A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually.

b. Control rod drop (scram) times for all four control rods shall be determined annually.

c. All control rods shall be visually inspected for deterioration quinquennially.

d. The transient (pulse) rod pneumatic cylinders and the associated air supply systems shall be inspected annually, and cleaned and lubricated if necessary.

e. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient (pulse) rod system shall be performed.

f. A channel test of each of the reactor safety system channels shall be performed prior to each day's operation or prior to each operation extending more than one day.

g. A channel verification of the settings of the seismic switch shall be performed annually or as soon as possible after observation or report of a seismic event of sufficient magnitude to trip the switch.

h. A calibration of the fuel temperature measuring channel shall be performed annually.

Basis. The control rods are inspected and drop times checked to assure safe scram operations. The surveillance intervals for those and the channel surveillances are selected based on the past history for over 40 years at this facility and are adequate to correct for long term drifts and other instrument problems.

[Note 1 : the seismic switch manufacturer makes no recommendation for recalibration and believes accelerometer settings remain effective for the life of the device.]

[Note 2 : the daily test of the seismic switch involves creating a simulated seismic event by tapping on the switch and initiating a reactor scram signal invoking a control rod scram. This establishes operational functionality and is more than a channel check.]

13.2 Maximum Hypothetical Accident (MHA) - Rupture of Single Fuel Element in Air.

13.2.1. Summary

The MHA for TRIGA® reactors has been adopted (NUREG-1537) to be the release of fission products from a single fuel element whose cladding has been stripped. This is analyzed both under water and in air. The assumption for the MHA is made that maximum fission product inventory has been achieved in the most active fuel element, i.e., that the reactor has been operating for an infinite time period prior to the release. This is obviously a highly conservative assumption for the UCI reactor which has operated on a very intermittent schedule since it was commissioned. The actual likelihood for such a complete rupture event is very small, but is not further considered here. This section examines the consequences of the postulated MHA. The most significant activity released from the perspective of personal exposure risk is that of iodine isotopes since this delivers exposure to the thyroid gland. Direct radiation is also considered for risk to any occupants of adjacent areas not subject to risk from the volatile materials.

The released radioactivity is the product of the fission product inventory and the release fraction. Consideration is then given to the exposure risk to personnel within the facility, those in surrounding spaces including laboratories or offices, and release beyond the building via the building ventilation system. Factors included are the facility volume, the ventilation purge rate and an estimated plating factor for iodine radio-nuclides on walls and facility structures which will reduce inhalation exposures.

The total number of fission product nuclei present from the fuel was estimated from Blomeke and Todd[1] and the release fraction was used from NUREG 1282. The modest levels predicted in the MHA come about as a result of the very low value of fission product release fraction established for TRIGA® fuel elements that have stripped their cladding.

13.2.2. Fission Product Inventory

The inventory of fission product gases, halogens and noble gases in the B-ring fuel element in which the average power density is 1.45 times greater than the core average, was determined by assuming infinite operation at 250 kw with [redacted] elements in the core.

Using the data developed by Blomeke and Todd[2] results in the volatile fission products compiled in Table 13-1.

Two groups of gaseous fission products are recognized. The first comprises bromine and iodine isotopes that will dissolve if water is present, and be airborne if it is not. The second group comprises the insoluble volatiles: krypton and xenon isotopes. The second are the major source of radioactivity in the room (and outside) if the unclad element were to be under water.

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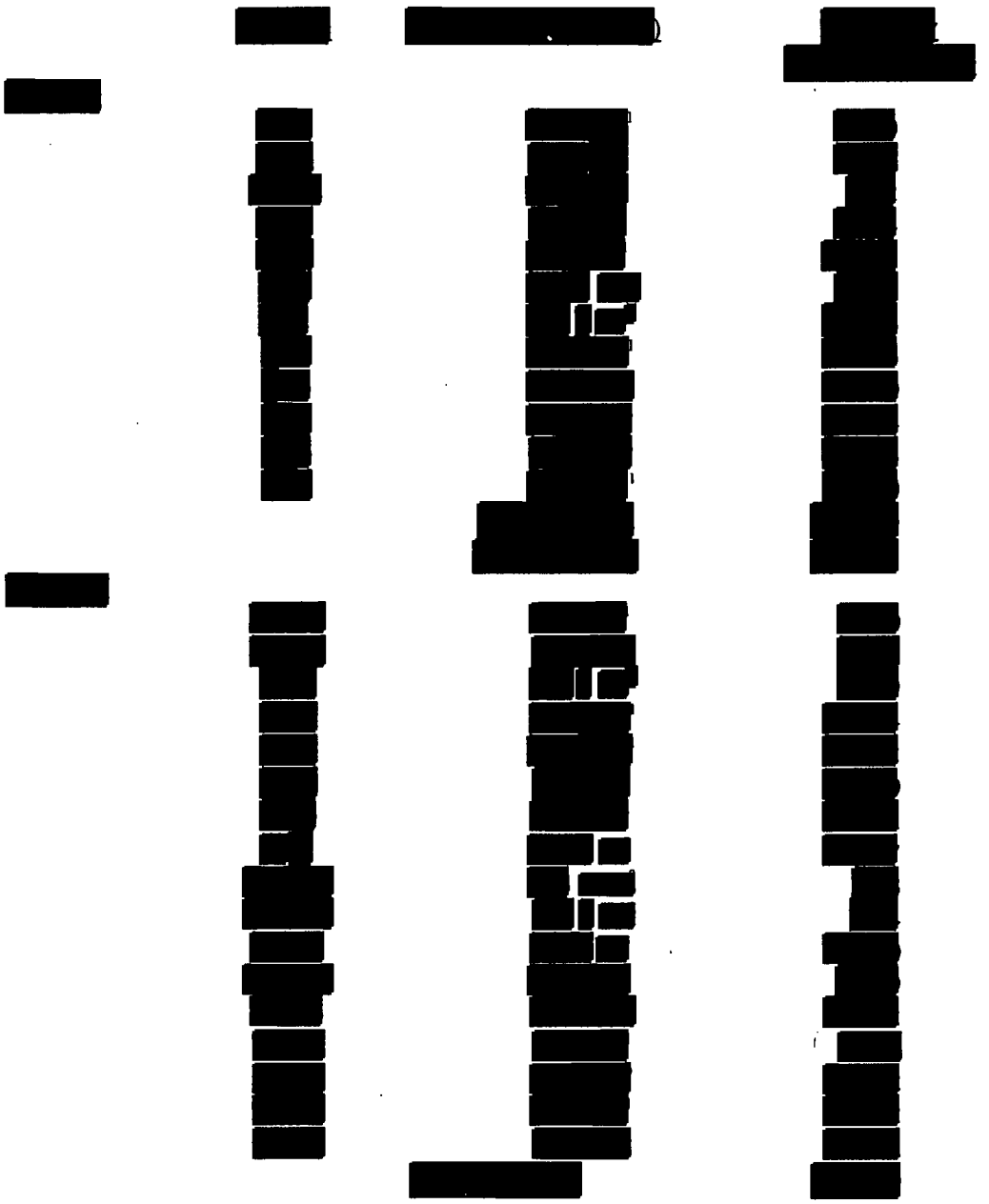
1 J. 0. Blomeke and Mary F. Todd, "U-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time", ORNL- 2127.

2 J. 0. Blomeke and Mary F. Todd, "U-235 Fission Product Production as a Function of the Thermal Neutron Flux, Irradiation Time, and Decay Time", ORNL-2127, Oak Ridge National Laboratory, August 1957-November 1958.

TABLE 13-1

GASEOUS FISSION PRODUCTS IN MAXIMUM POWER FUEL ELEMENT 250 kw SATURATED

ACTIVITY-[redacted]



13.2.2.2 Fission Product Release Fraction

The effect of a fuel element rupture resulting in a release of fission products from a B - ring fuel element was evaluated in the 1968 SAR. More recently the issue of fission product release fraction for TRIGA® fuel[3] was reviewed. The value proposed remains the same, so that the earlier analysis remains valid. The potential release is the product of the Fission Product Inventory and the Release Fraction.

NUREG-1282 quotes a release fraction of 1×10^{-5} that has been measured for 8.5% uranium TRIGA® fuel. Some indications in experiments are that the release increases with fuel temperature. As a result, facilities have adopted

$$release\ fraction = 1.5 \times 10^{-5}$$

as an appropriate value for hypothetical use.

Multiplying this release fraction by the inventory of gaseous fission products produced in the maximum power fuel element, as given in Table 13-1, gives the total activity that would be released should the integrity of a fuel element cladding be compromised. These values are shown in Table 13-2

Table 13-2 Releases from Maximum Power Fuel Element Unclad in Air

	Total Curies	Total Released (Curies)
Noble Gases	[redacted]	[redacted]
Iodine	[redacted]	[redacted]
Halogens	[redacted]	[redacted]

13.2.3 Pool Water Activity if Rupture is Under Water.

We review briefly the course of events if the rupture did occur in water, in which case the soluble (Group I) halogen products will remain in the water, totaling [redacted] curies, or [redacted] microcuries [redacted]

Since the volume of water in the reactor pool is [redacted] cm^3, the activity concentration is [redacted] µc/cm^3 ([redacted] Bq/mL) In 24 hours the activity would decrease to [redacted] µc/cm^3 [redacted] Bq/mL). The activity remains moderately high because of the small decay constants for I-129, I-131, and I-133. The demineralizer can be used to safely concentrate these for subsequent disposal. 20 millicuries of activity on the resin is a readily handled quantity by trained personnel. Thus the personnel exposure consequences of such an incident are relatively minor.

13.2.4 Predicted Exposures to Personnel Inside the Reactor Room if Rupture in Air.

[3] USNRC, Safety Evaluation Report on High Uranium Content, Low Enriched Uranium-Zirconium Hydride Fuels for TRIGA® Reactor, NUREG-1282, August 1987

13.2.4.1 Direct External Personnel Exposure.

The maximum exposure to a person in the reactor room will occur if the fission products from a ruptured element are distributed within the room and no air change occurs. The key parameter is the room air volume in which the fission products will be mixed. This is calculated to be 6.6 x 10^8 cm^3.

The total release of noble gases is [redacted] curies, or [redacted] microcuries and the volume of the reactor room is 6.6 x 10^8 cm^3 so that the concentration on mixing is [redacted] microcuries/cm^3, or [redacted] Bq/cm^3.

If this is assumed to be an infinite cloud around a person for dose rate calculation, then the formula[4] to use for external beta and gamma ray dose equivalent rate is

$$\dot{H} = 0.26\, S\, E_\gamma \; + \; 0.23\, S\, E_\beta \text{ rem/sec}$$

Where:

S = source strength in C_i/cm^3 = [redacted] C_i/cm^3.

E_γ = mean gamma ray energy in MeV = 1.0 MeV assumed (see table 11.3, ref 4)

E_β = mean beta energy in MeV = 0.4 MeV assumed (see table 11.3, ref 4)

The result is 1.7 x 10^{-11} Rem/sec. More conveniently this is converted to mr/h by multiplying by 1000 x 3600 or 3.6 x 10^6, yielding a prediction of a negligible dose equivalent rate of 6 x 10^{-5} mr/h. Including the halogens increases this to 1 x 10^{-4} mr/h, less than 1 microrem/h. The fact that this exposure is so small is attributed to the low operating power, an extremely small fission product release fraction, and the large volume of the reactor area for dilution. If, in the hypothesized accident, the water were to be absent from the pool, the volume for dilution would be increased by a further 10%, reducing the exposure still further. However, with the water present, halogens are likely to be absorbed rapidly into the water leaving only the noble gases in the source term.

The controlling factor for personnel exposure will be a longer term dose to the thyroid from the halogen activity, specifically iodine.

13.2.4.2. Thyroid Exposure.

The decay rate of halogens ([redacted] Bq/mL) will accumulate rapidly on the Continuous Air Monitor (CAM) filter which has a flow rate of 30L/min, or 500 mL/sec. Thus in 1second, [redacted] Bq should be collected. Assuming a typical counting efficiency of 9%, the count rate would already be 8100 cpm which will alarm the instrument (alarm is set to 5,000 cpm), switch to emergency purge ventilation, and indicate the room should be evacuated.

Once the emergency purge exhaust is operating, venting air at the rate of at least 240 cfm, the dose rate to any person remaining in the reactor room will decrease.

The dose to the thyroid can be calculated at any time (t) from[5]

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[4] J. R. Lamarsh, and A.J. Baratta, "Introduction to Nuclear Engineering", 3rd edition, Prentice hall, New Jersey, 2001, page 654-656.

[5] J. J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, 1962.

$$D_t = \frac{(5.92\times10^2)A_\tau f_a \bar{E}\left[1 - e^{-\lambda_e t}\right]}{m\lambda_e} \text{ (rads)}$$

where, A_τ = inhaled iodine, in curies

fa = fraction which is deposited in critical organ

$\bar{E}$ = effective energy absorbed by thyroid per disintegration (Mev)

λ_e = effective decay constant, comprising both radioactive decay and biological elimination

m = mass of the thyroid in grams

in a long time, $t = \infty$, and using T_e ($= 0.693/\lambda_e$), the equation becomes:

$$D_\infty = \frac{(8.54\times10^2)A_\tau f_a \bar{E} T_e}{m} \text{ (rads)}$$

$$= \text{constant} \times \sum_i A_{\tau i} \bar{E}_i T_{ei}$$

where the parameters are for the ith isotope.

The inhaled iodine is given by

$$A_{\tau i} = R_\tau \bullet \frac{A_i}{v}$$

where $\frac{A_i}{v}$ is the activity concentration, and R the breathing

rate for time τ. If τ is made equal to 1 sec, then

$$D_\infty = \frac{(8.54\times10^2)f_a R}{m} \sum_i \frac{A_i E_i T_{ei}}{v} \text{ (rads/sec)}$$

For the standard man[6] $f_a = 0.23$

$m = 20$ grams

$R = 10 m^3/8$ hr

$= 3.47 \times 10^{-4}\ m^3 sec^{-1}$

and the value of the constant is thus 3.41×10^{-3}.

The data necessary for the summation are in the cited reference[7], and the necessary activity

[6] "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)," Health Physics 3, June 1960.

[7] J. J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, 1962.

concentrations are calculated from Table 13-1 using 1.5×10^{-5} for the release fraction. The summation for I-131 through I-136 yields a value for $D_\infty = 0.36$ rad/sec.

In Oregon State University's SAR, they adopted a method which suggests halogen release plates out to reduce the airborne concentrations to less than 25% of the initial release.
With this assumption, the revised dose rate expected to the thyroid in the unlikely event of a fuel element cladding rupture in air, is 0.25 x 0.36 rads/second = 0.09 rads/second. If the water in the pool is assumed to be present there is a high likelihood that this will rapidly absorb the majority of halogens from the air and reduce the source for thyroid exposure to an individual in the facility to even less than 25% of that released.
Given a 2 minute evacuation time for personnel from the facility, the revised maximum dose estimate is 11 Rads to the thyroid.

Given that 50 rem to the thyroid is a limiting annual exposure[8] a person will have a longer time than 2 minutes to evacuate. This is a very conservative estimate since in addition to the comments made above that would reduce exposure, infinite operation at full power prior to the accident was assumed. Nor was credit taken for operation of the room emergency purge exhaust which will also act to reduce the dose received.

13.2.5 Predicted Exposure Outside the Building

On detection of release of radioactive fission products into the reactor room, the continuous air monitor will close the normal exhaust and start the emergency purge. The room air will then be exhausted through a filter at 240 cfm (0.113 m^3/sec) which will not remove the noble gases and may not remove all the iodine isotopes released. However, credit may also be taken for plate-out on surfaces of the iodine isotopes.

The ventilation system (SAR Section 3.6) discharges the emergency purge exhaust into the main building exhaust which has three exhaust fans designed to create a flow of 3 x 80,000 cfm, further diluted to 3 x 128,000 cfm at a height of 120 feet above the building roof by means of Axijet® structures. From that point, natural dispersion, aided by the jet stream, will further dilute the effluents.

From the above, it is reasonable to assume a conservative dilution of the exhaust effluent by a factor of $384,000/240 = 1.6 \times 10^3$ Using this, and assuming no climatic or other dilution outside the building exhaust flow, the maximum dose to an individual's thyroid may be calculated.

The calculation assumes:

(1) Complete mixing in the reactor room at all times

(2) The person is immersed in the effluent from the building exhaust flow in which the concentration of radioactivity X_t at any time t is equal to $C_t/1.6 \times 10^3$ where C_t is the concentration in the exhaust and in the room.

With (1) C_t is given by a simple first order rate law: $C_t = C_0 \exp\left[-\frac{a}{V}t\right]$

where a = exhaust rate in m^3/sec = 0.113 m^3/sec

V = room volume in $m^3 = 6.6 \times 10^2$ m^3

[8] 10CFR 20.1201(1)(i)

C_o = initial concentration, at t = 0

the inhaled activity $A_{dt} = R.\ X_t.\ dt$ as above and thus the total amount inhaled up to time t is

$$A_t = \frac{RC_o}{1600}\int_o^t \exp\left[-\frac{a}{V}t\right]dt$$

$$= \frac{RC_oV}{1600a}\left[1-\exp(-\frac{a}{V}t)\right]$$

If the exposure is for an infinite time,

$$A_\infty = RC_oV/a.1600$$

and, comparing with the previous calculation, the total dose

$$\text{int } D_\infty = \frac{D_\infty(rads/\text{sec})\bullet V(m^3)}{1600\bullet a(m^3/\text{sec})} = \frac{D_\infty(rads/\text{sec})\ x\ 6.6x10^2}{1600\ x\ 0.113}$$

$$= 0.33 \text{ rad}$$

If the person were to remain in the effluent cloud for only one hour, the corresponding figure is less than 0.2 rad. Thus we may conclude that for a location above the roof of the building in completely calm conditions, the maximum possible dose to the thyroid from a fission product release will be significant. Since the building of the roof is not open to the public, the exhaust plume is over 100 feet above the roof level, and is approximately 100 feet above ground level, the general public would be unable to approach to within over 100 feet of the initial source cloud before it becomes further diluted by random or wind-driven plume expansion and dilution.

Gulf General Atomic has developed computer programs which take allowance both for decay and for further meteorological dilution.[9] Calculations using these programs for a similar situation to that presented here[10] indicate that these factors will reduce the maximum dose to a person a short distance away from the building by several orders of magnitude. Thus for a slow leak rate from a typical room ($a = 1.69 \times 10^{-2}$ m^3/sec) and very stable atmospheric conditions (Pasqual F), they found predicted exposures of:

	Whole Body γ (millirem)	Thyroid (millirem)
Outside Max dose (12 hours) (General Public)	2.9×10^{-3}	2.6
Inside Max dose (1 hour) Occupational Personnel	6.0×10^{-3}	7,000

Similar calculations using more sophisticated programming for MNRC[11] resulted in exposure

[9] GA-6511, "GAD∅SE and GA∅SET--Programs to Calculate Environmental Consequences of Radioactivity Release". GAMD-5939, "PHICHK and PHIBAR programs to Calculate Atmospheric Dispersion".

[10] Addendum to GA-6499, "Hazards Report for the Oregon State University 250 kw TRIGA Mark II reactor--dated June 23, 1965", April 20, 1966.

[11] SAR, McClellan Nuclear Radiation Facillity, 1999, Chapter 13.

predictions within similar ranges when corrected for their greater source term by roughly a factor of 8. However the UCI greater room size and expanded building ventilation system reduce the calculated exposure rates significantly.

The high variability of local climatic conditions actually prevalent when an accident occurs make it relatively ineffective to try to predict accident exposures with any greater degree of certainty. It seems very likely that the present predictions will be conservative, and that accidents of this hypothetically very serious nature at this facility will not result in unacceptably hazardous exposures either to personnel in the facility, nor to the general public. Since the probability of such an event could increase during fuel handling operations, in order to further guard against the likelihood of public exposures, operational regulations at UCINRF preclude the presence of any inessential or untrained personnel within the facility whenever operations such as fuel handling are in progress. Events involving handling of irradiated fuel elements in air are subject to even greater restrictions and personnel training, and include procedures to exclude the public from all areas adjacent to the facility. This was the practice followed during the transfer of one cask load of irradiated fuel elements into the facility in 1974. The protocol for that procedure is on file for further use and inspection.

13.2.6 Predicted Exposure to Rowland Hall Occupants Near the Facility

The calculations presented in section 13.2.4.1. can be applied to assessment of potential exposure in an MHA to individuals located in adjacent areas outside the reactor facility.

There is no pathway for airborne activities to reach the adjacent areas of the building, so that the deep dose rates established in that section for individuals within the facility may be utilized. Such individuals will not be immersed in an infinite cloud of radioactive volatiles, so the facility may be considered as a single source. Further only gamma-ray exposures would be experienced since the walls and windows are thick enough to absorb all beta radiation. There are no doors directly opening from the facility to adjacent building areas, so no leakage to interior building areas is anticipated.

Section 13.2.4.1 established a uniform dose rate within the facility of less than 1 microrem/hour. This rate from within the facility would represent no hazard to those in adjacent rooms and hallways, even assuming no distance factors or absorption reductions.