SECTION VI . SAFETY ANALYSIS
of the
FINAL SAFETY ANALYSIS REPORT

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VI. SAFETY ANALYSIS

A. Potential Consequences Associated with TRIGA Reactor Operations

The potential radiological consequences associated with routine and non-routine operation of the TRIGA Mark F Reactor have been reviewed herein. This section presents the results of this study.

Routine operational risks are present in the manipulation of any radioactive material because of its potential for biological or physical damage. This study investigated potential consequences which may be associated with routine operation, including the handling of radioactive materials, reactor power transients, experiments associated with reactor operations, and production of radioactive gases (primarily through activation of argon).

Non-routine operational risks may arise from the improper handling or malfunctions of materials and equipment. The specific problems investigated were fuel element cladding failure, radioactive contamination of the reactor shielding water, loss of shielding and cooling water, and Design Basis Accidents.

In order to evaluate the potential consequences associated with the operation of this reactor the following assumptions have been made:

1. The reactor will normally have two modes of operation:

   Steady State Operation at power levels not to exceed 1.0 megawatt (MW).[1] For calculations, the assumption of operation in the steady-state mode at a power level of 1.0 MW was utilized. The total reactor operation has been less than 400 MW-hr from June 1964 to September 1980. The total reactor operation from January 1980 to September 1980 has been less than 26 MW-hr.

   Pulsed Operation achieved by intentionally placing the reactor on a prompt critical excursion by making a step insertion of reactivity above critical utilizing the transient rod and appropriate scrams and interlocks. The maximum step insertion is limited by the technical specifications to 2.8% Δ k/k reactivity in the pulse mode.[1] The

average maximum pulse used during AFRRI's TRIGA pulse operation has been 28 MW-sec. For calculations, the assumption of a maximum excursion pulse of 40 MW-sec was utilized as an upper limit for pulse operations or an inadvertant transient.

2. The reactor operations will be supervised by individuals who are qualified operators trained in the detection and evaluation of radiological consequences.

In summary, the consequences from the Design Basis Accidents of a fuel elements drop accident or a fuel element clad failure accident were insignificant. The maximum calculated on-site whole body dose was less than 2 mrad beyond 25 meters from the AFRRI facility. The maximum calculated on-site thyroid dose was less than 57 mrad beyond 25 meters from the AFRRI facility. Doses from these postulated accidents to individual beyond the boundary of the National Naval Medical Center site would be significantly less than 1 mrad. Therefore, it is concluded that the operation of the AFRRI reactor in the manner authorized by Facility License No. R-84 does not represent an undue risk to the health and safety of the operational personnel or the general public.

B. Routine Operational Risks

B.1 Handling Radioactive Material

Because of the potential biological effects of radioactive material, special precautions should be taken in the handling of these materials. Reactor operations must be supervised by responsible individuals who are trained in the detection and evaluation of radiological consequences. Administrative, operational, and health physics procedures will be followed, and special equipment and procedures which are needed to maintain the ALARA (as low as is reasonably achievable) concept of radiation protection will be used.

The radiological consequences associated with fuel elements are of the same nature as that associated with isotope production. Because of their high radiation levels, fuel elements should be kept under water for shielding purposes. Administrative policy requires that a fuel element not be removed from the reactor pool for at least two (2) weeks following its use in the reactor. When the fuel element is removed from the reactor tank, a conventional fuel element transfer cask may be used to reduce radiation levels to within acceptable limits. A Design Basis Accident (Section C) is a fuel element drop during such a transfer in which a cladding failure occurs in air.

When proper administrative, operating and health physics procedures are utilized, the handling of radioactive materials does not represent a significant risk to the health and safety of operating personnel or the general public.

B.2 Reactor Power Transients

The following discussion is based on experimentation and testing performed by General Atomic and presented in Reference 2. Theoretical estimates are also reviewed. The U-ZrH fuel elements used in the TRIGA reactor are capable of operating under conditions of transient experiments for delivery of high intensity bursts of neutrons. Fuel elements with 8.45 wt.% U have operated repeatedly in General Atomic's Advanced TRIGA Prototype Reactor (ATPR) to peak power levels of over 8,000 MW providing neutron fluence per pulse of approximately 1.0E+15 nvt (neutrons/unit area).

The ATPR fuel elements were subjected to thousands of pulses of 2,000 MW or more. The inherent safety of the fuel element stems from the prompt negative temperature coefficient of reactivity, which causes the automatic termination of a power excursion before any core damage results. This temperature coefficient has been measured to be approximately $0.017 reactivity loss per 1°C rise in fuel temperature.

The reactor loading is limited by the technical specifications to a maximum of 3.5% Δ k/k ($5.00) excess reactivity above cold critical, with or without all experiments in place.[1] Thus, the maximum reactivity transient that could possibly occur would be that produced by the rapid insertion of the entire available amount of reactivity. Experiments at General Atomic's ATPR of step insertions of up to 3.5% Δ k/k ($5.00) reactivity have been performed. The fuel elements were subjected to thousands of pulses of 2,000 MW or more and attained temperatures of up to 1000°C.

Theoretical estimates based on the Fuchs-Nordheim mathematical model of the AFRRI-TRIGA reactor have also been made. It was shown that an insertion of 2.8% Δ k/k (AFRRI-TRIGA technical specification limit) results in a maximum average fuel temperature of less than 550°C.

Therefore, based on operating experience of the ATPR and the Fuchs-Nordhein matehmatical model, it can be concluded that the rapid insertion of the total excess reactivity of 3.5% Δ k/k would not represent an undue risk to the operating personnel or the general public.

B.3 Improper Fuel Loading

Fuel loading of the reactor is always supervised by trained, licensed, supervisory personnel. All reactor monitoring and shutdown devices will be activated during loading. The worst possible case of improper fuel loading would be for the operator to insert a fuel element in a core already slightly critical. In a core near its critical point all of the inner fuel positions would be occupied so that the extra fuel elements could be added only in a peripheral position, where a fuel element worth is approximately 0.24% Δ k/k. Since step additions of 2.1% Δ k/k excess reactivity are made on a routine basis for pulsing the reactor, the addition of 0.24% Δ k/k would not present a danger of damaging the reactor or fuel. The reactor would undergo a mild transient and then operate at a steady state power level of about 60 KW.

Even in the extremely unlikely event that a fuel element in the "B" ring should be improperly handled, its rapid insertion would result in an addition of about 0.83% Δ k/k. As indicated above, such an addition would not result in any damage to the reactor or the fuel and would not represent an undue risk to the health and safety of the operating personnel or the general public.

B.4 Production of Radioactive Gases in the Reactor Coolant

The production of radioactive gases by the reactor in its associated facilities originates through neutron activation of elements in the air or water. One of the most important of these activation products is radioactive argon (Ar-41) with a half-life of 1.83 hours.[3] Calculations are based on a temperature of 70°F (21°C), and Ar-40 content of air of 0.94% by volume.

In the calculation to determine the amount of argon dissolved in the reactor pool water, it is assumed that the argon follows Henry's Law. At a water temperature of 70°F (21°C), the corresponding water vapor pressure is 26 mm Hg. The partial pressure for air would therefore be 760-26 = 734 mm Hg. Using an argon content of air as 0.94% by volume the resulting partial pressure of argon is 734x(9.4E-3) = 7 mm Hg. Applying Henry's Law, a saturation concentration of argon in water of 1.367 E-8 gram mole of Ar-40 per 1 cm^3 H_2O is obtained. Using a thermal neutron absorption cross section of 6.1E-25 cm^2 for Ar-40[3], the macroscopic absorption cross section of argon in water becomes 5E-9 cm^{-1}.

Activity production, assuming saturation condition and an irradiation time, t, can be given as:

$$A_p = \frac{\Phi_n (1-e^{-\lambda t_p}) \Sigma_a}{3.7\ E+4} \qquad (1)$$

Where:

A_p = activity in reactor water ($\mu Ci/cm^3$)
Φ_n = thermal neutron flux (n/cm^2-sec)
t_p = core circulation time (sec)
Σ_a = macroscopic absorption cross section (cm^{-1})
3.7 E+4 = constant to convert disintegration per second to microcuries
λ = decay constant (sec^{-1})

The average thermal neutron flux (Φ_n) in the reactor core is estimated to be 1.0 E+13 n/cm^2-sec at the 1.0 MW power level. The water circulates in the core by natural convection and is estimated to change completely in approximately 4 seconds. The Ar-41 decay constant is 1.05×10^{-4} sec^{-1}. The core of the reactor holds 3.4E+4 cm^3 of water and the rate of flow of water through the core is 0.9E+4 cm^3/sec.[4] Substituting the appropriate values in equation 1 yields:

A_p = 5.7E-4 $\mu Ci/cm^3$,

and a total core activity of:

Q = 5.1 μCi/sec.

The travel time of Ar-41 from the core to the water surface, a minimum distance of 457.2 cm (15 ft), has been estimated to be 42 seconds. Assuming the decay to be negligible, the maximum activity reaching the surface is 5.1 μCi/sec.

Under saturated, steady state conditions, (a highly conservative assumption for this facility), the maximum rate at which Ar-41 can escape from the water surface will be 5.1 μCi/sec, and an equivalent amount of Ar-40 will dissolve in the water to replace the Ar-41 depletion. As water temperature increases, the water vapor pressure will increase and the amount of dissolved Ar-40 will decrease, resulting in a lower generation of Ar-41 escaping to the atmosphere.

The radioactive argon (Ar-41) will escape from the reactor pool, dissipate in the reactor room air, and recirculate through the ventilation system. It is estimated that this system handles 2.80E+6 cm^3 (92,000 ft^3 of air), and that it

recirculates 3.34E+8 cm^3/min (11,800 ft^3/min) through the system and exhausts its share 8.24E+7 cm^3/min (2910 ft^3/min) of the 1.133E+9 cm^3/min (40,000 ft^3/min) thru the stack.[4] The added exaust is from the other AFRRI facilities ventilation systems.

Since the air is recirculated through the system at a high rate, it is assumed that the Ar-41 is distributed throughout the areas handled by the system and that the Ar-41 decay is negligible. If it is assumed that there is continuous reactor operating time, equilibrium conditions can be assumed.

Since equilibrium conditions are assumed, the same concentration of Ar-41 will exist in the reactor room as in the exhaust air. The concentration in the reactor room is:

$$\frac{5.1 \text{ Ci/sec}}{1.37 \text{ E+6 } cm^3/\text{sec}} = 3.7 \text{ E-6 } \text{Ci}/cm^3$$

The concentration in the stack is 2.7 E-7 $\mu Ci/cm^3$ and is further dispersed in the air before reaching the surrounding population. The gamma dose rate in the reactor room can be estimated for the Ar-41 concentration of 3.7 E-6 $\mu Ci/cm^3$ by assuming submersion in a spherical source equivalent to the reactor room volume. The exposure rate can be given as:

$$D = \frac{S_v}{K}(1-e^{\mu r}) \qquad (2)$$

Where: D = exposure rate (r/hr)

S_v = volume source (Mev/sec-cm^3)

μ = attenuation coefficient for air (cm^{-1})

K = flux to dose conversion factor (Mev/cm^2-sec per R/hr)

r = radius of equivalent sphere (cm)

The volume of the reactor room is approximately 9.2 E+8cm^3 which is equivalent to a sphere with a radius (r) of 600 cm.[4] An Ar-41 concentration of 3.7 E-6 $\mu Ci/cm^3$ has a gamma source strength (S_v) equal to 0.08 Mev/cm^3-sec. For the Ar-41 gamma photon of 1.29 Mev, the attenuation coefficient for air (μ) is 7.2 E-5 cm^{-1} and the flux to dose conversion factor (K) is 5.44 E+5 Mev/cm^2-sec per R/hr. Substituting these values into equation 2 results in an exposure rate from the calculated Ar-41 concentration of 0.18 mR/hr. Such an exposure rate is more than an order of magnitude below the equivalent allowable quarterly exposure rate of 2.5 mrem/hr in a restricted area.

Another important activation product is radioactive nitrogen (N-16) with a half-life of 7.14 seconds. The activation occurs as a result of the oxygen content in the water from the O^{16} (n,p) N^{16} production process. The concentration of oxygen in water is approximately 0.89 gram mole of O-16 per $1cm^3$ H_2O. Using a thermal neutron absorption cross section of 1.85 E-29 cm^2 for O-16[5], the macroscopic absorption cross section of oxygen in water becomes 9.9E-6 cm^{-1}.

Substituting the appropriate values in equation 1 yields:

$Ap = 8.4$ E+2 $\mu Ci/cm^3$,

and a total core activity of:

$\dot{Q} = 7.6$ Ci/sec.

Using a travel time for N-16 from the core to the water surface of 42 seconds, the maximum activity reaching the surface is 0.13 Ci/sec.

The radioactive nitrogen will escape from the reactor pool, dissipate and decay rapidly (7.14 second half-life) in the reactor room air. Assuming the volume of air above the surface of the reactor pool in which the N-16 activity per second is distributed with negligible decay is 100 cm, the concentration of N-16 activity would be approximately 1.2 E-2 $\mu Ci/cm^3$ with a gamma source strength (Sv) equal to 2.1 E+3 Mev/cm^3-sec ($E\gamma$ = 4.6 Mev/disintegration). For the N-16 primary gamma photon of 6.13 Mev, the attenuation coefficient for air (μ) is 3.2 E-5 cm^{-1} and the flux to dose conversion factor (K) is 8.4 E+5 Mev/cm^2-sec per R/hr. Substituting these value into equation 2 for an equivalent sphere of 136 cm results in an exposure rate from the calculated N-16 concentration of 350 mR/hr immediately above the reactor pool.

Based upon measurements made during reactor operations at 1MW power level, the typical Ar-41 production rate has been approximately 0.5 $\mu Ci/sec$ as compared to the calculated value of 5.1 $\mu Ci/sec$, a factor of 10 reduction. A typical concentration at the stack has been approximately 3 E-8 $\mu Ci/cm^3$ as compared to the calculated value of 2.7 E-7 $\mu Ci/cm^3$, a factor of 10 reduction. Gamma radiation levels over the pool ranged from 20 mR/hr to 75 mR/hr as compared to the calculated value for N-16 activity of 350 mR/hr, approximately a factor of 5 reduction. Gamma radiation levels around the reactor pool chain ranged up to 14 mR/hr, indicating the rapid reduction due to N-16 decay as it dissipated into the reactor room volume.

B.5 Experiments

All experiments performed as part of the TRIGA reactor operations are reviewed by the Reactor and Radiation Facility Safety Committee and must be authorized prior to their performance. The technical specifications contain requirements that must be met by the experiment before such experiments can be performed in the TRIGA reactor. Experiments are always supervised by trained, licensed supervisory personnel. However, failure of an experiment is possible and worst case conditions can be calculated to determine the postulated consequences.

The two worst case conditions for failure of an experiment could result in instantaneous insertion of reactivity or the release of radioactive material from an experiment undergoing activation in the reactor. For an experiment failure in which reactivity could be added, the worst possible case would be the prompt addition of less than 0.36% Δ k/k in either Exposure Room 1 or 2. As discussed for the case of improper fuel loading (Section B.3), the addition of 0.36% Δ k/k would be within the range of an improper fuel loading condition. Such an addition would not result in any damage to the reactor or the fuel.

For an experiment failure in which radioactive material could be released from the experiment; i.e., activation products, the worst case would be the prompt release of the radioactive material to the atmosphere. The quantity of radioactive isotopes produced will not cause the inventory of any specific isotope to exceed that authorized by the AFRRI by-product license. An authorized experiment involves the irradiation of 20 liters of argon gas for 1 hour at a power level of 1 MW. The resulting activation would result in a total Ar-41 activity of 5.6 Ci in the sealed container. If the container should fail and release all of the Ar-41 activity, the resulting total whole body dose would be less than 1 mrad to an individual beyond 75 meters from the AFRRI facility. The failure of this authorized experiment represents the worst case for radiological consequences from an experiment failure in the AFRRI TRIGA reactor. Such a whole body exposure would not represent an undue risk to the health and safety of the operating personnel or the general public.

C. Non-Routine Operational Risks

C.1 Radioactive Contamination of Reactor Shielding Water

Contaminant material susceptible to neutron irradiation in the shield water is maintained at low concentrations by the water purification system. The conse-

quences associated with a failure of the fuel-element cladding and subsequent fission product contamination of the water have been calculated and studied experimentally as described in Section C.2.a. The results show that in the improbable event of fuel element cladding failure, the water can be decontaminated by the resins of the purification system. Manufacturing inspection and quality controls assure that the possibility of cladding failure is minimal.

Experiments conducted over a period of 11 years by General Atomic[2] on 8.5% U-ZrH fuel elements under various conditions have shown that a small fraction of fission products are released from U-ZrH fuel into the gap between fuel and cladding. This release fraction varies from less than 1.5E-5 for an irradiation temperature of 350°C to approximately 1.0E-2 at 800°C.

Three mechanisms (recoil, diffusion, and dissolution) have been shown to be involved in fission product release. The first mechanism is the fission fragment recoil into the gap between the fuel and clad. This effect predominates at temperatures up to approximately 400°C. In this range the recoil release rate is dependent on the fuel surface-to-volume ratio but independent of temperature. This is the most important mechanism in this system and is considered in the calculations dealing with fission product release following fuel element cladding failure (Section C.2). At temperatures above approximately 400°C, the diffusion-like process predominates. The amount released is dependent on the fuel temperature, the fuel surface-to-volume ratio, the length of time of irradiation, and the isotope half-life. This mechanism is of little significance to the fission-product release fraction considered in this system. The dissolution of unclad fuel in water, steam, and air have been shown to be relatively low at temperatures up to 600°C.[2]

C.2 Fuel Element Cladding Failure

a. Summary of Previous Experience

Since the original development of the TRIGA reactor, General Atomic has maintained a program of research and development oriented toward continuing improvement in the design of the fuel elements so that TRIGA reactors could be operated in both steady-state and pulsing modes without undue concern regarding the integrity of the fuel.

Extensive operational testing of TRIGA fuel elements has been conducted by General Atomic in the Torrey Pines TRIGA and the prototype TRIGA Mark F since

the early 1960's. Experiments on fuel with 8.5 wt% U were conducted over a period of 11 years under a variety of conditions. These experiments include:[2]

- 1960 - The measurement of the quantity of a single fission product isotope release from a full-size TRIGA fuel element during irradiation.
- 1966 - The measurement of the fractional release of several isotopes from small specimens of TRIGA fuel material during and after irradiation at temperatures ranging from approximately 25°C to 1100°C.
- 1971 - The measurement of the quantities of several fission product isotopes released from a full-size TRIGA fuel element during irradiation in a duplication of the 1960 experiment.

In the course of developing a reactor for pulsed operation, General Atomic has experienced three fuel element cladding failures in their Torrey Pines TRIGA reactor. These cladding failures have been of two types:

1. Two cladding failures caused by the thermal ratcheting of the aluminum cladding during the cycling.
2. A cladding failure associated with an element instrumented with internal thermocouples.

In the original TRIGA fuel element design, ratcheting occurred during pulsed operations of the reactor. Modification of the fuel element essentially eliminated this cause of failure. The present cladding is stainless steel and of greater durability than the original aluminum which experienced the metal clad failures. The cladding failure of an instrumented element was apparently the result of internal pressure built up, in-seepage water or possibly the pressure of volatile material inside the element.

The consequences of the cladding failures were all minor and in the worst case experienced may be summarized as follows:

1. The activity in the reactor water reached a maximum of 0.2 $\mu Ci/cm^3$. It decayed very rapidly and was measured 24 hours after the cladding failure to be 5×10^{-5} $\mu Ci/cm^3$, a 4000-fold reduction.

2. The activity in the air of the reactor room reached about ten times the maximum permissible concentration for fission products and then decayed rapidly. The faulty fuel element was removed and experiments were resumed 2 hours after the activity release. The maximum integrated exposure to operating personnel resulting from this release was 1 mr.

3. The noble gases were not collected on the air monitor filters used, but it may be inferred from the nature of the particulates collected that only noble gas fission products escaped from the TRIGA pool in significant proportions when the fuel cladding failed.

It is concluded from these experiments that a cladding failure or even the simultaneous failure of the cladding of several fuel elements would not constitute an undue risk to the health and safety of the operating personnel or the general public.

b. Calculation of Maximum Fission Product Release After a Fuel Element Cladding Failure

Calculations and a related experiment have been made to determine the maximum concentration of fission products that might be present in the reactor room air following a fuel element cladding failure.

The calculations are based on the fact that as the reactor operates, the fission products will build in the uranium-zirconium fuel mixture until a equilibrium concentration is reached for each nuclide. The resulting equilibrium nuclide concentration of fission products depends upon the total energy release in the reactor, the decay process for each nuclide, and the yield of the species from fission. Only the kryptons, the xenons and the iodines will migrate into the gap between the fuel material and fuel element cladding. A Design Basis Accident (Section C.4) is a fuel element cladding failure during a pulse operation or inadvertent transient following steady state operation.

b.1 Calculated Fission Product Inventory

To determine the various inventories of the fission products produced in the core, data was used from Reference 6 for a steady state power level of 1 MW. The inventories for the kryptons, xenons, and iodines were calculated as shown in Table 1. The prompt fission product inventory for a 40 MW-sec pulse operation

TABLE 1

Gaseous Fission Product Inventory
1 MW Steady State Operation

Isotope	Half-Life*[6] ($T_{1/2}$)	Fission Yield[6] (%)	Saturation Activity (Ci)	Gamma Source Strength[3] (Mev/Sec)
Krypton (Kr)				
83	1.86 h	0.52	4.5 E+3	7.0 E+12
85 m	4.40 h	1.30	1.1 E+4	1.9 E+14
85	10.76 y	0.271	2.4 E+3	1.8 E+11
87	76.0 m	2.53	2.1 E+4	1.1 E+15
88	2.80 h	3.56	3.1 E+4	2.0 E+15
89	3.20 m	4.59	4.0 E+4	3.4 E+15
TOTAL		12.77	1.1 E+5	6.7 E+15
Xenon (Xe)				
131 m	11.9 d	0.017	1.5 E+2	8.8 E+11
133 m	2.30 d	0.16	1.4 E+3	1.2 E+12
133	5.27 d	6.69	5.8 E+4	1.8 E+14
135 m	15.6 m	1.8	1.6 E+4	3.1 E+14
135	9.2 h	6.3	5.5 E+4	5.0 E+14
137	3.8 m	6.0	5.2 E+4	2.9 E+15
138	14.0 m	5.9	5.1 E+5	2.4 E+15
TOTAL		26.87	2.3 E+5	6.3 E+15
Iodine (I)				
131	8.05 d	2.91	2.5 E+4	3.4 E+14
132	2.28 d	4.26	3.7 E+4	3.2 E+15
133	21.0 h	6.69	5.8 E+4	1.0 E+15
134	52.0 m	7.80	6.7 E+4	4.8 E+15
135	6.7 h	6.17	5.4 E+5	3.6 E+15
TOTAL		27.83	2.4 E+5	1.3 E+16
GASEOUS TOTAL		67.47	5.8 E+5	2.6 E+16

* y = year, d = day, h = hour, m = minute

was calculated using the data from Reference 6 with the build-up and decay of activities from Reference 7 for prompt fission. The prompt inventories for the kryptons, xenons and iodines as calculated are shown in Table 2.

These inventories will be utilized in the calculation of radiological consequences due to the various fuel element cladding failures which may result during the operation of the AFRRI-TRIGA reactor.

b.2 Determination of Fission Products In Gap[2]

In order to determine the actual percentage of fission-product gases that escape from the fuel material and collect in the gap between the cladding and the fuel material, experiments were conducted in the TRIGA reactor at General Atomic. A fuel element was fabricated with a sealed tube that vented the gap to a charcoal-filled cold trap at the surface of the reactor tank. All of the fission-product gases that accumulated in the gap were collected in the liquid-air-cooled charcoal trap by purging the system with helium, and the trap was analyzed. This measured amount of radioactive noble gases enabled the determination of the fraction of the fission products that diffused through the uranium-zirconium hydride material into the gap.

Although the measured amount of radioactive noble gases for the operating conditions in the AFRRI reactor fuel would indicate a gap activity percentage of less than 1 E-2 percent, the theoretical limit of 1 E-1 percent gap activity for fission product gases of noble gases and iodines as states in Reference 2 will be used in the consequence analysis for the Design Basis Accidents (Section C.4).

b.3 Air and Water Activity Following Cladding Failure

As shown in Table 1, the total quantity of all gaseous fission products in a TRIGA core for 1 MW operation at equilibrium is 5.8 E+5 curies. For the purposes of this calculation, it is assumed that the fractions of the iodine, krypton, and xenon isotopes produced that collect in the gap between the fuel material and fuel cladding are 0.1 percent.[2] Thus, the total core activity in the gap is calculated to be 5.8 E+2 curies. The maximum amount of fission products that could be released in the event of a cladding failure of a single average fuel element for the 85 fuel element AFRRI-TRIGA core is less than 7 curies during steady state operation. As shown in Table 2, the total quantity of all gaseous fission products from a 40 MW-sec pulse operation is 1 E+4 curies in the core with a gap activity of 10 curies. The additional release in the event of a cladding failure in an average fuel

TABLE 2

Gaseous Fission Product Inventory
for 40 MW-sec Pulse Operation

Isotope	Half-Life*[6] (T½)	Fission Yield[6] (%)	One-Minute Decay Activity[7] (Ci)	Gamma Source Strength[3] (Mev/Sec)
Krypton (Kr)				
85 m	4.4 h	1.3	2.9	5.0 E+10
87	76.0 m	2.53	87.0	4.4 E+12
88	2.8 h	3.56	91.0	6.0 E+12
89	3.2 m	4.59	3600.0	3.1 E+14
TOTAL		11.98	3.78 E+3	3.2 E+14
Xenon (Xe)				
135 m	15.6 m	1.8	56.0	1.1 E+12
135	9.2 h	6.3	1.4	1.4 E+10
137	3.8 m	6.0	4600.0	2.6 E+14
138	14.0 m	5.9	1300.0	6.0 E+13
TOTAL		20.0	5.96 E+3	3.2 E+14
Iodine (I)				
132	2.28 d	4.26	5.7	4.7 E+11
133	21.0 h	6.69	4.3	7.4 E+10
134	52.0 m	7.80	165.0	1.2 E+13
135	6.7 h	6.17	57.0	3.7 E+12
TOTAL		24.92	2.3 E+2	1.6 E+13
GASEOUS TOTAL		56.9	1.0 E+4	6.6 E+14

* -d = day, h = hour, m = minute

element is less than 0.2 curies. The total release from a cladding failure event is approximately 7 curies from an average fuel element during pulse operation following steady state operation.

As given in Reference 4, the volume of water in the TRIGA reactor tank is approximately 5.7 E+7 cm^3 and the volume of air in the reactor room is approximately 9.2E+8 cm^3. For the purpose of calculation, it is assumed that all the gaseous fission products in the gap are available for release. It is assumed that 99.8% of the iodine dissolves in the water with the remaining iodine, krypton, and xenon released into the reactor room atmosphere. Of the 7 curies that could be released in the clad failure of an average fuel element, less than 3 curies would be radioiodine while the remaining 4 curies would be the noble gases, krypton, and xenon. Therefore, the concentration in the water would be less than 0.05 $\mu Ci/cm^3$, while the air concentration would be less than 0.005 $\mu Ci/cm^3$.

b.4 Calculation of Doses Following Cladding Failure

Since krypton and xenon are inert gases, the exposure due to their presence in air is from submersion. To calculate the exposure rate, equation 2 for submersion exposure from a spherical sources is used. The gamma source strengths for the kryptons, xenons, and iodines are given in Table 1 for the steady state operation and Table 2 for the pulse operation. As previously determined, the volume of the reactor room is approximately 9.2 E+8 cm^3 which is equivalent to a sphere with a radius (R) of 600 cm. The gamma source strength from kryptons and xenons for steady-state operation is calculated as 1.3 E+16 Mev/sec while for pulse operation it is calculated as 0.1 E+16 Mev/sec or a total of 1.4 E+16 Mev/sec. Using the gap activity as 0.1 percent and an 85 fuel element core, the clad failure of an average fuel element will release less than 1.7 E+11 Mev/sec of kryptons and xenons into the reactor room volume. The air concentration of 0.005 $\mu Ci/cm^3$ previously calculated is equivalent to a gamma volume source strength (S_v) of $2x10^2 Mev/cm^3$ -sec.

For the noble gas mixture, the attenuation coefficient for air (μ) is 9.7 E-5 cm^{-1} and the flux to dose conversion factor (K) is $5x10^5$ Mev/cm^2-sec per R/hr assuming a 0.7 Mev gamma photon. Substituting these values into equation 2 results in an exposure rate from the calculated noble gas concentration of about 0.2 rem/hr. Within an hour, the exposure rate will be reduced to less than 0.1 rem/hr.

Based upon these conservative calculations for a fuel element cladding failure during pulse operation following steady-state operation, a person could remain in the reactor room for more than 10 hours before exceeding the allowable quarterly occupational exposure limit of 1.25 rem.

Standard operating procedures require prompt evacuation of the reactor room of reactor personnel following an indication of excessive airborne radioactivity in the reactor room. Reentry to the reactor room would be determined by the Physicist-in-Charge (PIC) when the airborne concentration was safe for normal operations to resume.

C.3 Loss of Shielding and Cooling Water

It is highly improbable that the reactor will lose the shielding and cooling water. Loss of this water can occur by only two means; the tank may be pumped dry, or a tank rupture may allow the water to escape in a relatively short period of time. However, should such an event occur during the operation of the reactor, the reactor is set to shut-down if: 1) the pool water level is 15 feet or less above the pool floor, and 2) the loss of moderation will stop the fission process.

In the event that the only mechanism for heat removal after the loss of coolant is natural convection of air through the core, the fuel element cladding would not fail and the fission products would be retained within the fuel element.[2]

In the event that a rupture should occur and repair of the reactor wall is required, the core and any stored fuel elements that are present would be moved behind the lead shielding doors, so repairs can be made safely.

The predicted gamma dose rates that would result from loss of pool water before the reactor core could be moved behind the lead shielding doors would be approximately 350 mR/hr near the reactor pool (10 feet) due to skyshine (air scatter) and 3 R/hr on the roof above the reactor due to direct gamma exposure. Predicted whole body dose rates to individuals within the AFRRI facility but outside the reactor room should be less than 10 mR/hr. Whole body dose rates to individuals outside the AFRRI facility beyond 10 meters should be less than 1mR/hr. These dose rates would be reduced by a factor of 4 within 1 hour due to decay of the fission products in the reactor core.

C.4 Design Basis Accidents

a. Fuel Element Drop Accident

A Design Basis Accident (DBA) for the AFRRI reactor is postulated to be the occurrence of a cladding failure of a fuel element after a two-week period where the saturated fission product inventory of a 1 MW steady state operation has been allowed to decay after being taken out of the operating core and placed in storage. The cladding failure could occur when the fuel element is withdrawn from the reactor pool. While the fuel element is exposed to air, a cladding failure could occur coincidentally or due to a drop. The probability that such an accident would occur is considered to be extremely remote. The probability of clad failure has been further reduced under the postulated accident conditions by the substitution of stainless steel clad for the former aluminum clad on the fuel elements. The fission products released from the gap will depend upon the temperature of the fuel following two weeks decay. This temperature is expected to be less than 100°C. The temperature needed to volatilize iodine (183°C) is, therefore, not reached and gaseous iodine should not be released. The kryptons and xenons will normally remain in the gaseous state and be released (100%). Although iodine will not be volatile under the assumed accident conditions, a release of 1% of the gap activity has been assumed for calculational purposes.

The diffusion factor (χ/Q) and finite cloud correction factors (F) are dependent upon the distance from the source and are presented in Table 3. These parameters were based upon a Pasquill type F stability condition with a 1m/sec wind speed and a cross-sectional area for the AFRRI facility of 450 m^2. The methodology described in Reference 8 was used to determine the diffusion factor. The methodology described in Reference 9 for defining the ratio of gamma dose from a finite cloud to an infinite cloud with the same centerline concentration was used to determine the finite cloud correction factor. Table 4 presents the source terms of radioactivity and thyroid and gamma source strengths in the total core for 1 MW steady state operation after two weeks of decay. These values are to be reduced to 0.1 % for the gap activity and for the 85 fuel element core to determine the amount of fission products that might be released in an average fuel element drop accident resulting in clad failure. The total release is less than 0.15 curies with a gamma source strength of 3.6 E+8 Mev/sec. The iodine release fraction is assumed to be 1 percent for a thyroid source strength of 1.3 E+3 rads.

TABLE 3

Diffusion Factor and
Finite Cloud Correction Factor
Used for Dose Calculations

Distance (m)	χ/Q[8] (sec/m^3)	F*[9]
25	1.0 E-1	1.5 E-2
50	2.7 E-2	2.7 E-2
75	1.2 E-2	4.2 E-2
100	7.5 E-3	5.5 E-2
150	3.7 E-3	8.0 E-2
200	2.2 E-3	1.0 E-1

*F represents a ratio of gamma doses from a finite size cloud to that calculated from an infinite cloud with same center-line concentration.

TABLE 4

Gaseous Fission Product Inventory
for 1 MW Steady State Operation
with 2-Week Decay

Isotope	Activity (Ci)	Gamma Source Strength (Mev/sec)	Thyroid Source Strength (rad)
Krypton (Kr)			
85	2.4 E+3	1.8 E+11	--
Xenon (Xe)			
131 m	67.0	3.9 E+11	--
133 m	21.0	1.8 E+10	--
133	9.2 E+3	2.9 E+13	--
TOTAL	9.3 E+3	2.9 E+13	--
Iodine (I)			
131	7.5 E+3	1.0 E+14	1.1 E+10
132	5.2 E+2	4.5 E+13	2.8 E+7
TOTAL	8.0 E+3	1.5 E+14	1.1 E+10
GASEOUS TOTAL	2.0 E+4	1.8 E+14	1.1 E+10

To calculate the whole body gamma dose to an individual outside the AFRRI facility, the following equation was utilized:

$$D_\gamma = 0.25\,(\bar{E}_\gamma)(\bar{Q})(\bar{\chi}/Q)(F) \qquad (3)$$

Where:

D_γ = integrated whole body dose (rad)
$\bar{E}_\gamma$ = gamma energy released per disintegration (Mev/dis)
$\bar{Q}$ = total activity released (Ci)
$\bar{\chi}/Q$ = diffusion coefficient (sec/m^3)
F = ratio of gamma dose from finite cloud to infinite cloud with same centerline concentration
.25 = constant for semi-infinite cloud dose analysis

To calculate the thyroid dose to an individual outside the AFRRI facility, the following equation was utilized:

$$D_{th} = (BR)(\bar{\chi}/Q)(S_{th}) \qquad (4)$$

where:

D_{th} = integrated thyroid dose (rad)
BR = individual breathing rate (m^3/sec)-3.47×10^{-4} m^3/sec
$\bar{\chi}/Q$ = diffusion coefficient (sec/m^3)
S_{th} = iodine inhalation source term (rad)

The whole body and thyroid doses calculated for individuals downwind from the AFRRI facility in the event of a fuel element drop accident are given in Table 5. The conservative assumption is made that the radioactive gases released in the AFRRI reactor room will be released directly to the atmosphere without significant holdup within the facility. The current design of the AFRRI reactor room would cause isolation of the reactor room by automatic closure of the ventilation pathway to the stack and would prevent excessive leakage to other parts of the AFRRI facility due to the air-tight access doors.

The calculated whole body dose is insignificant at all distances downwind from the AFRRI facility. The maximum calculated on-site thyroid dose of 45 mrad

TABLE 5

Calculated Doses to Individuals Standing Downwind of AFRRI Facility Following Fuel Element Drop Accident

Distance (m)	Whole Body Dose (mrads)	Thyroid Dose (mrads)
25	0.0004	45
50	--*	12
75	--	6
100	--	3
150	--	2
200	--	1

*Insignificant

is more then a factor of 100 below the projected thyroid dose of 5 rad, below which no protective action is recommended.[10] The calculated thyroid dose for individuals beyond the boundary of the National Naval Medical Center (NNMC) site within which the AFRRI facility is located would be significantly less than 1 mrad.

b. Fuel Element Clad Failure Accident

Another Design Basis Accident (DBA) for the AFRRI reactor is postulated to be the cladding failure of a fuel element during a pulse operation or inadvertent transient following steady state operation at 1 MW. The assumed fission product inventories in the core are those given previously in Tables 1 (1 MW operation) and Table 2 (40 MW-sec pulse operation) and analyzed in Section C.2. As discussed for the failure of cladding in a fuel element drop accident, the likelihood of occurrence of a fuel element clad failure during normal operation or an inadvertent transient is considered to be extremely remote. The data given in Table 3 as well as equations 3 and 4 will be used in the evaluation of this DBA to calculate the whole body and thyroid doses to individuals downwind from the AFRRI facility.

Table 6 presents the source terms of radioactivity and thyroid and gamma source strengths postulated to be released to the atmosphere from the AFRRI reactor room. The resulting whole body and thyroid doses calculated for individuals downwind from the AFRRI facility are given in Table 7. The same conservative assumption made for the fuel element drop accident regarding the prompt release of radioactive material to the atmosphere has been made for this DBA.

The calculated whole body dose is insignificant at all distances downwind from the AFRRI facility. The maximum calculated on-site whole body dose is less than 2 mrad or a factor of 500 below the projected whole body dose of 1 rad, below which no protective action is recommended.[10] The maximum calculated on-site thyroid dose of 57 mrad is nearly a factor of 90 below the projected thyroid dose of 5 rad, below which no protective action is recommended.[10] The calculated thyroid dose for individuals beyond the boundary of the NNMC site would be significantly less than 1 mrad.

TABLE 6

Gaseous Fission Product Inventory Released to Atmosphere
Following Fuel Element Clad Failure Accident

Isotope	Activity (Ci)	Gamma Source Strength (Mev/sec)	Thyroid Source Strength (rad)
Krypton	1.4	3.6 E+10	--
Xenon	2.9	7.9 E+10	--
Iodine	0.06	3.1 E+8	1.7 E+3
TOTAL	4.4	1.7 E+11	1.7 E+3

TABLE 7

Calculated Doses to Individuals Standing Downwind of AFRRI Facility Following Fuel Element Clad Failure Accident

Distance (m)	Whole Body Dose (mrads)	Thyroid Dose (mrads)
25	1.7	57
50	0.8	16
75	0.6	7
100	0.4	5
150	0.3	3
200	0.2	2

D. References

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