# APPENDIX B

# RADIOLOGICAL IMPACT OF ACCIDENTS

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# APPENDIX B

## RADIOLOGICAL IMPACT OF ACCIDENTS

### B.1 Maximum Hypothetical Accident (MHA)

Numerous safety committees that have reviewed TRIGA® reactor operations have considered potential accidents including rapid insertion of reactivity, loss of heat removal, loss of coolant, metal-water reactions, rearrangement of fuel, fuel aging, and accidents during handling of irradiated fuel. Chapter 13 of this document discusses such accidents. This appendix addresses the consequences of the accepted maximum hypothetical accident (previously called the design basis accident) for a TRIGA® reactor: a cladding rupture of one fuel element with no decay and subsequent instantaneous release of fission products into the air. This is commonly referred to as a single element failure in air. This is also the most severe of all accidents for TRIGA® reactors and is analyzed to examine potential radiation doses to the reactor staff and the general public.

At some point in the lifetime of the MNRC reactor, used fuel within the core may be moved to new positions or removed from the core. Fuel elements are moved only during periods when the reactor is shut down. The most serious fuel-handling accidents involve spent or used fuel that has been removed from the core and then dropped or otherwise damaged, causing a breach of the fuel element cladding and a release of fission products. As noted previously, the standard or accepted maximum hypothetical accident for TRIGA® reactors involves failure of the cladding of a single fuel element after extended reactor operations, no time to decay, and subsequent release of the fission products directly into the air of the reactor room. While a credible scenario for this accident is hard to establish, it will be assumed that such an event can take place and does so immediately after reactor operation with a fuel element that has been run at full power (1 MW) for a period of 1 year. This operating history is very conservative in nature as it is unlikely the facility is ever operated more than 2,000 MWhrs in a year for the remainder of the facility's life. The intent of the fuel element failure in water during 1 MW (maximum steady-state power) accident to provide a better understanding of the radiological consequences of a more realistic accident than the MHA. The intent of the fuel element failure in water 24 hours after shutdown is to provide a better understanding of the radiological consequences of damaging a fuel element while it is being handles remotely in the reactor tank.

The fission product source term is calculated by hand based on the highest fission rate fuel element in the LCC of 17.69 kW. This element would theoretically operate at the highest temperature and have the highest fission inventory. Fission product yield and half-life data were taken from published IAEA data. Very short half-lived isotopes (i.e. less than 1 minute) were excluded from the source term due to the general lack of dose conversion factors and their low expected contribution to radiological consequences. To help compensate for this underestimation a 25% overestimation is made in all other fission product inventories.

The fission product inventory at shutdown is listed in Table B-1. The data are for the volatile fission products contained in a fuel element run to saturation at the highest core power density.

For both accidents being analyzed in this appendix, a release fraction of 2.36 x $10^{-5}$ is assumed for the release of noble gases and halogens from the fuel to the cladding gap. This release fraction is developed in Chapter 4 and is based on the maximum measured fuel temperature (400 C) which highest average fuel temperature expected operating at 1 MW. This temperature is likely an overestimate given the measured (via instrument fuel element) of the hottest element is closer to 330 C. This thermal output is the considered to be the maximum that can be achieved by the LCC. In addition, for the accident where the cladding failure occurs in air, it is very conservatively assumed that 25% of the halogens released to the cladding gap are eventually available for release from the reactor room to the outside environment. A release fraction of 2.36 x $10^{-5}$ was also assumed for a single fuel element failure 24 hours after shutdown, even though the element would be at ambient temperatures, in order to keep the radiological consequences conservative.

**Table B-1 Source Terms for One-Element Accident for MHA**

| Group I | | Group II | |
|---|---|---|---|
| Nuclide | Activity (Ci) | Nuclide | Activity (Ci) |
| Br-83 | 113.4 | Kr-83M | 113.4 |
| Br-84 | 208.4 | Kr-85M | 272.9 |
| Br-84M | 35.3 | Kr-85 | 3.7 |
| Br-85 | 272.9 | Kr-87 | 541.6 |
| I-131 | 611.4 | Kr-88 | 751.0 |
| I-132 | 911.8 | Kr-89 | 954.1 |
| I-133 | 1417.4 | Xe-131M | 8.6 |
| I-134 | 1656.5 | Xe-133M | 40.0 |
| I-134m | 77.0 | Xe-133 | 1417.4 |
| I-135 | 1328.6 | Xe-135M | 232.7 |
| | | Xe-135 | 1383.6 |
| | | Xe-137 | 1296.8 |
| | | Xe-138 | 1332.8 |
| **TOTAL** | **6632.7** | | **8348.6** |

**Table B-2 Source Terms for One-Element Accident for 24-hour decay**

| Group I | | Group II | |
|---|---|---|---|
| Nuclide | Activity (Ci) | Nuclide | Activity (Ci) |
| Br-83 | 0.11 | Kr-83M | 0.015 |
| Br-84 | 0 | Kr-85M | 6.6 |
| Br-84M | 0 | Kr-85 | 3.7 |
| Br-85 | 0 | Kr-87 | 0.001 |
| I-131 | 561 | Kr-88 | 2.15 |
| I-132 | 0.62 | Kr-89 | 0 |
| I-133 | 637 | Xe-131M | 8.1 |
| I-134 | 0 | Xe-133M | 29.1 |
| I-134m | 0 | Xe-133 | 1242 |
| I-135 | 105.6 | Xe-135M | 0 |
| | | Xe-135 | 222 |
| | | Xe-137 | 0 |
| | | Xe-138 | 0 |
| **TOTAL** | **1304.3** | | **1513.7** |

This value is based on historical usage and recommendations from References B.2, B.3, B.4, B.5, and B.6, where Reference B.2 recommends a 50% release of the halogens. References B.3 and B.4 apply a natural reduction factor of 50% due to plate out in the building. This latter 50% applied to the 50% of the inventory released from the fuel element cladding gap results in 25% of the available halogen inventory reaching the outside environment. However, this value appears to be quite conservative based on the 1.7% gap release fraction for halogens quoted in References B.7 and B.8.

For the single element accident in water, it is conservatively assumed that most of the halogens released from the cladding gap remain in the water and are removed by the demineralizer system. However, a small fraction, approximately 2.5% of the total halogens released to the cladding gap are, in this case, assumed to escape from the reactor tank water into the reactor room air, which is more conservative than assuming total (100%) solubility of the halogens as is sometimes done for TRIGA® reactors (Reference B.18). However, even assuming 2.5% halogen release from the pool water will almost certainly result in an overestimate of the actual radioiodine activity released into the room because of the use of a 50% halogen gap release fraction rather than the 1.7% documented in References B.7 and B.8. In addition, about 50% of the halogens released from the water are expected to plate out in the reactor building before reaching the outside environment (References B.3 and B.4).

The experience at TMI-2, along with some experiments, indicate that the 50% halogen release fraction is much too large. Smaller releases, possibly as little as 0.6% of the iodine reaching the cladding gap may be released into the reactor room air due in part to a large amount of the elemental iodine reacting with cesium to form CsI, a compound much less volatile and more water soluble than elemental iodine (Reference B.8).

The Lawrence Livermore National Laboratory program “HotSpot” was used to determine the worst case radiation dose impact to the public for a variety of atmospheric stability classes. Over the past several years the program has become the industry standard for relatively simple Gaussian plume modeling of radioactive material release.

Furthermore, it was assumed that all of the fission products were released to the unrestricted area instantaneously, which would maximize the dose rate to persons exposed to the plume during the accident. It is also assumed that the release height of the radioactive material to be the height of the floor of the reactor room (19 feet above ground level) not the 60 feet MNRC stack. This was done to keep the radiological results as conservative as possible. Additionally, a very calm wind speed of 1 m/s was selected as lower wind speeds in these types of releases produce greater radiological consequences. Dose conversion factors from FGR 11 (ICRP 30 series) were used to along with a breathing rate of 3.3e-4 $m^3/s$ to calculated various doses to the receptors of interest (1.0 m above ground level). Calculations were done for Pasquill weather classifications A through F. All other inputs not mentioned above were left on their default settings in Hotspot.

Worker dose was calculated by assuming an instantaneous release of the radioactive source term into the reactor room and not being removed by the ventilation system. This was done to keep the radiological consequences conservative. The reactor room volume was used to calculate the individual derived air concentration (DAC) values for each isotope of interest. The various DAC values and egress times in the reactor room were used to calculate the final worker dose consequences. Note the reactor room was assumed to large enough to mimic a semi-infinite cloud for the external radiation sources (primarily radio-xenon). As the reactor room is much smaller than a semi-infinite cloud this method will over predict expected worker dose from external radiation.

1. Initial Fission Product Source;

    a. Reactor Power level - 1 MW;

    b. Operating time - 365 days;

2. Release Fractions;

    a. For the purposes of the maximum hypothetical accident and the accident where the pool water remains present, it was assumed that at the time of fuel cladding failure, a fraction of the radionuclide in the inventory given in Table B-1 was instantaneously released into the air of the reactor room. In one scenario this instantaneous release occurs directly into the reactor room air, while in the other projected accident the pathway to the air requires migration through the pool water which will reduce the halogen release. Also, for the halogens, a further reduction in activity is expected to occur due to plateout in the building. Thus, the fraction ($w_i$) of the fission product inventory released from a single fuel element which reaches the reactor room air and then the atmosphere in the unrestricted environment outside the facility will be as follows:

$$w_i = e_i \times f_i \times g_i \times h_i \, ;$$

where:

$e_i$ = the fraction released from the fuel to the fuel-cladding gap;

$f_i$ = the fraction released from the fuel-cladding gap to the pool if water is present, or directly to the reactor room air if no water is present;

$g_i$ = the fraction released from the pool water to the reactor room air; and

$h_i$ = the fraction released from the reactor room air to the outside (unrestricted) environment.

b. Based on information provided previously in this appendix, $e_i$ was set at 2.38 x $10^{-5}$ for both noble gases and halogens and at zero for other fission products. The values of $f_i$, $g_i$, and $h_i$ for the two accident scenarios being considered are given in Table B-3.

**Table B-3. Values of Release Fraction Components**

| Fission Product | $f_i$ | | $g_i$ | | $h_i$ |
|---|---|---|---|---|---|
| | w/o pool water | w/ pool water | w/o pool water | w/ pool water | |
| Noble Gases | 1.0 | 1.0 | N/A | 1.0 | 1.0 |
| Halogens | 0.5* | 0.5* | N/A | 0.05 | 0.5 |

Offsite Dose: As can be seen below in table B-4 the offsite radiological consequences are low, which is consistent with the MHA of similar TRIGA reactors, primarily due to high retention of radionuclides by the fuel itself during such an accident. The maximum expected CDE for the thyroid and TEDE outside of the MNRC fence line occur under atmospheric D at a distance of 66 m from the MNRC reactor room. The thyroid CED is 11 mrem and the TEDE is 0.56 mrem, both well below the 10 CFR 20 annual limits for the public. The evaluation distances selected for table B-4 were based on the closest point from the MNRC reactor room the site boundary (fence line), the closest inhabited building, and the closest residence.

**Table B-4 Radiation Doses to Members of the General Public Under Different Atmospheric Conditions and at Different Distances from the MNRC Following a Fuel Element Cladding Failure in Air (MHA). All doses in mrem, CDE - Committed Dose Equivalent, TEDE - Total Effective Dose Equivalent**

33 meters (MNRC Fence Line)

| Weather Class | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| CDE Thyroid | 9.1 | 11 | 8.3 | 3.6 | <0.01 | <0.01 |
| TEDE | 0.46 | 0.56 | 0.41 | 0.18 | <0.01 | <0.01 |

93 meters (Closest Inhabited Building)

| Weather Class | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| CDE Thyroid | 1.6 | 3.3 | 6.3 | 9.4 | 6.7 | 0.40 |
| TEDE | 0.08 | 0.16 | 0.31 | 0.46 | 0.33 | 0.02 |

480 meters (Closest Residence)

| Weather Class | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| CDE Thyroid | 0.06 | 0.14 | 0.32 | 0.72 | 1.7 | 4.1 |
| TEDE | <0.01 | <0.01 | 0.01 | 0.03 | 0.08 | 0.18 |

Maximum Dose and Distance

| Weather Class | A | B | C | D | E | F |
|---|---|---|---|---|---|---|
| CDE Thyroid | 14 (@19m) | 11 (@32m) | 11 (@48m) | 11 (@66m) | 8.3 (@130m) | 7.1(@240m) |
| TEDE | 0.69 (@19m) | 0.56 (@32m) | 0.55 (@48m) | 0.56 (@66m) | 0.40 (@130m) | 0.33 (@240m) |

**Table B-5 Atmospheric Modeling Values Based on Hotspot**

| Atmospheric Stability Classification | Sigma y (m) | Sigma z (m) | $\chi_{max}$ ($\mu$Ci/cm$^3$) | Distance to Max Dose (m) | Maximum TEDE Dose (mrem) |
|---|---|---|---|---|---|
| Very Unstable (A) | 4.18 | 3.80 | $1.69*10^{-3}$ | 19 | 0.69 |
| Moderately Unstable (B) | 5.11 | 3.84 | $1.40*10^{-3}$ | 32 | 0.56 |
| Slightly Unstable (C) | 5.27 | 3.82 | $1.35*10^{-3}$ | 48 | 0.55 |
| Neutral (D) | 5.26 | 3.78 | $1.34*10^{-3}$ | 66 | 0.56 |
| Slightly Stable (E) | 7.75 | 3.83 | $9.2*10^{-4}$ | 130 | 0.40 |
| Moderately Stable (F) | 9.49 | 3.71 | $7.2*10^{-4}$ | 240 | 0.33 |

Worker Dose:

As indicated by the results in Table B-5, the occupational dose to workers who evacuate the reactor room within 5 minutes following the MHA should be approximately 300 millirem TEDE and 2,495 millirem Committed Dose Equivalent to the thyroid. If evacuation occurs within 2 minutes, as it no doubt will because the reactor room is small and easy to exit, the doses drop to 120 millirem TEDE and 998 millirem CDE. All of these doses are well within the NRC limits for occupational exposure as stated in 10 CFR 20.

**Table B-6 Worker dose for MHA**

| | CDE Thyroid (millirem) | TEDE (millirem) |
|---|---|---|
| 2 minute room occupancy | 998 | 120 |
| 5 minute room occupancy | 2495 | 300 |

Furthermore, if the pool water remains in place and there is 24 hours of decay, the 5 minute and 2 minute occupational doses (TEDEs) drop to 8.4 and 21.1 millirem, respectively.

Offsite radiological consequences will be much less significant than the MHA given the significantly smaller source term and that for this release the source term would likely be effluenced through the stack and not directly from the reactor room elevation.

**Table B-7 Radiation Dose Worker Following an Element Failure in Water with 24 hours of Decay**

| | CDE Thyroid (millirem) | TEDE (millirem) |
|---|---|---|
| 2 minute room occupancy | 75.3 | 8.4 |
| 5 minute room occupancy | 188 | 21.1 |

B.2 Radiation Dose Rate from the Core Following a Loss of Coolant Accident

B.2.1 Introduction

Even though there is a very remote possibility that the primary coolant and reactor shielding water will be totally lost, direct and scattered dose rates from an uncovered core following 1 MW operations have been calculated. Dose rates were calculated by constructing a simple, yet conservative, MNCP model of the MNRC reactor and facility. The calculation of activity is given below. Each radioactive decay is assumed to produce a single 1 MeV gamma-ray. The MCNP model utilizes a homogenized reactor (26 cm in height by 38 cm in diameter) to simplify the geometry greatly while approximating for some degree of self-shielding that will take place within uranium of the reactor. The 9 foot concrete shielding that constitutes the base of the reactor tank and the minimum 9 foot concrete cylinder surround the reactor tank were modeled. The 6 inch thick concrete walls of the reactor room were modeled however the roof of the facility was not modeled because it is essentially a thin steel corrugated structure that would provide little shielding or scatter. A 1-inch-thick iron disk 1 meter in diameter was placed 3 meters above the top of the reactor tank to simulate the MNRC "bridge" where the control drives are located. This disk was used to simulate the scatter that would take place during this event that would contribute to the dose rate of individuals not directly located in direct line-of-sight with the reactor. No other structural materials were included in the model, such as walls and ceilings. This was done to minimize the attenuation of scattered gamma-ray to yield conservative dose estimates. Attenuation and scatter in air were allowed for in this model.

Dose rates were calculated for an individual directly over the core standing in the reactor room, inside the reactor room but not in direct line-of-sight of the core, just outside the reactor room, inside the control, at the MNRC fence line, in the closes building, and the closest inhabited building.

The LOCA scenario:

Total Core Activity:

The total fission product activity as a function of time after shutdown was determined using the standard equation below (Reference B.16):

$$\text{Activity at time } \tau = 1.4\, P_o[(\tau - T_o)^{-0.2} - \tau^{-0.2}] \text{ Ci ;}$$

where:

$P_o$ = Thermal power (W);

$\tau$ = Time after reactor startup (d);

$T_o$ = Time reactor operating at power $P_o$ (d);

hence:

$\tau - T_o$ = Time after reactor shutdown (d).

The MNRC will operate a maximum 365 MWd per year; therefore, the operating profile used in the above equations was 8760 hours (365 days) at the full power of 1 MW. Increasing the operating time further makes little difference to the total fission product activity. The resultant fission product activity and source term can be seen in Table B-7.

| **Table B-8 Total Fission Product Activity After Shutdown** | | |
|---|---|---|
| Time After Shutdown | Total Activity (Ci) | Source Strength ($\gamma s^{-1}$) |
| 1 hour | $2.74 \times 10^{6}$ | $1.01 \times 10^{17}$ |
| 1 day | $9.70 \times 10^{5}$ | $3.59 \times 10^{16}$ |
| 1 week | $5.20 \times 10^{5}$ | $1.92 \times 10^{16}$ |
| 1 month | $2.86 \times 10^{5}$ | $1.06 \times 10^{16}$ |

B.2.2 Dose Rate Directly Above the Core

The results are given in Table B-8 and are in general agreement with results for the 2 MW Torrey Pines TRIGA® Reactor (Reference B.18) scaled to 1.0 MW MNRC operations.

| **Table B-9 Dose Rates on the MNRC Reactor Top After a Loss of Pool Water Accident Following 1 MW Operation** | |
|---|---|
| Time After Shutdown | Effective Dose Equivalent Rate (rem/h) |
| 1 hour | $1.68 \times 10^{3}$ |
| 1 day | $5.94 \times 10^{2}$ |
| 1 week | $3.18 \times 10^{2}$ |
| 1 month | $1.74 \times 10^{2}$ |

B.2.3 Dose Rate From Scattered Radiation in Reactor Room

The results below are given for a position located just inside the MNRC reactor room 2 meters from the edge of the reactor tank. This location was selected to be far enough away from the reactor tank so that there is no line-of-sight to the exposed core.

The results of the calculations for the scattered radiation dose rates can be seen in Table B-9.

**Table B-10 Scattered Radiation Dose Rates in the MNRC Reactor Room After a Loss of Pool Water Accident Following 1 MW Operation**

| Time After Shutdown | Effective Dose Equivalent Rate (mrem/h) |
|---|---|
| 1 hour | 235 |
| 1 day | 83.9 |
| 1 week | 46.1 |
| 1 month | 24.5 |

B.2.4 Dose Rate at Other Locations After a Loss of Pool Water Accident

The dose rates after a loss of pool water accident are provided in this section.

A position just outside (2 meters away) from the entrance to the reactor room is given to provide a worst-case estimate of personnel dose (Table B-10) in the MNRC equipment room or on the first level of the MNRC room. Should this accident scenario occur personnel could need to enter these areas to establish water to the reactor tank.

**Table B-11 Scattered Radiation Dose Rates in the MNRC Outside Reactor Room After a Loss of Pool Water Accident Following 1 MW Operation**

| Time After Shutdown | Effective Dose Equivalent Rate (mrem/h) |
|---|---|
| 1 hour | 89.8 |
| 1 day | 32.1 |
| 1 week | 17.7 |
| 1 month | 9.4 |

The MNRC control room is located at ground level approximately 14 m east of the core. The dose estimates below (Table B-11) are conservative as no shielding credit is taken for the roof and walls of the reactor facility.

| Table B-12 Scattered Radiation Dose Rates in the MNRC Control Room After a Loss of Pool Water Accident Following 1 MW Operation | |
|---|---|
| Time After Shutdown | Effective Dose Equivalent Rate (mrem/h) |
| 1 hour | 17.5 |
| 1 day | 6.2 |
| 1 week | 3.4 |
| 1 month | 1.8 |

The shortest distance to the MNRC fence line is 28 meters. The dose estimate given below (Table B-12) should be considered the highest possible doses possible for the public in this scenario.

| Table B-13 Scattered Radiation Dose Rates in the MNRC Fence Line After a Loss of Pool Water Accident Following 1 MW Operation | |
|---|---|
| Time After Shutdown | Effective Dose Equivalent Rate (mrem/h) |
| 1 hour | 16.1 |
| 1 day | 5.7 |
| 1 week | 3.0 |
| 1 month | 1.6 |

The closest building to the MNRC is an old industrial x-ray facility utilized by the Air Force. It is located 56 meters to the south of MNRC and has be uninhabited for nearly 20 years. No credit for building shielding is taken and the dose rates are presented below (Table B-13).

| Table B-14 Scattered Radiation Dose Rates in Closest Public (not inhabited) Building After a Loss of Pool Water Accident Following 1 MW Operation | |
|---|---|
| Time After Shutdown | Effective Dose Equivalent Rate (mrem/h) |
| 1 hour | 8.4 |
| 1 day | 3.0 |
| 1 week | 1.6 |
| 1 month | 0.85 |

The closest inhabited building to the MNRC is a large conference center. It is located 94 meters to the north east of MNRC and is regularly used. No credit for building shielding is taken and the dose rates are presented below (Table B-13).

**Table B-15 Scattered Radiation Dose Rates in Closest Public (inhabited) Building After a Loss of Pool Water Accident Following 1 MW Operation**

| Time After Shutdown | Effective Dose Equivalent Rate (mrem/h) |
|---|---|
| 1 hour | 3.5 |
| 1 day | 1.2 |
| 1 week | 0.69 |
| 1 month | 0.36 |

B.2.5 Mitigation of LOCA skyshine by Reflooding MNRC Reactor Core:

Though the probability of a complete LOCA and the resulting uncovering of the core is a very low and poses no threat of fuel overheating, the potential doses to the public can exceed 100 mrem (approximately 10 hours after the core becomes uncovered). Doses to the public may also eventually exceed 500 mrem, though this would require a member of the public to stand at the MNRC fence line for nearly 100 hours straight. Therefore, a method to reflood the core to reduce the dose to the public to levels below 10 CFR 20 limits is needed.

The LOCA event postulated in Chapter 13 will drain the 7,000 gallon reactor tank in one hour. The system previously required as an emergency core cooling system (to prevent fuel overheating after a complete LOCA for 2.0 MW operation) will be maintained in order to reflood the core to limit skyshine doses. The core reflooding system is based on a 1.35 inch diameter pipe that can be attached to the city water supply. Water from this system is emptied directly on to the core from just above the two-foot-high chimney structure. The system is initiated by a reactor operator connecting a fire hose from the city water supply to a core inlet pipe located just outside of the reactor room. The system requires approximately 15 minutes to initiate. A reactor operator would have ample time from receiving a low tank level alarm to when the core would become uncovered (~ 1 hour after the low tank level alarm) to begin reflooding the core.

In the event of the postulated LOCA the primary coolant would fill the void between the reactor tank the concrete monolith the tank sits in. From there the primary water would begin to flood the subterranean room below bay 4 (referred to bay 5). It is important to note that this system cannot provide flow greater than the postulated leak rate of the LOCA accident. Therefore, the reactor core can only be covered with water after bay 5 has been filled with water as well. Once bay 5 has been completely filled with water, the reactor core would be covered with approximately 4 feet of water. This would reduce the skyshine doses by a factor of 100 to 1000 and essentially end the skyshine hazard from the reactor.

Bay 5 is an excavated concrete lined room 24 feet wide by 27 feet long by 14 feet in depth. This corresponds to a volume of 8,064 cubic feet or 60,320 gallons. Based on a water pressure of 20 PSI and utilizing Bernoulli’s equation, a flow rate of 43 gallons per minute is expected for the core reflooding system. This corresponds to a flood time of bay 5 (and recovering the core) of just under 24 hours. Below in table B-16 are the 24 hour integrated doses for the postulated LOCA and reflooding the core event at various location. Public doses to the can theoretically surpass 100 mrem but cannot surpass 500 mrem.

| **Table B-16 Integrated Scattered Radiation Doses for Various Location After 24 hours and Reactor Core has been Reflooded** | |
|---|---|
| Location | Effective Dose Equivalent (mrem) |
| Control Room | 208 |
| Fence Line | 192 |
| Closest Building (not inhabited) | 100 |
| Closest Building (inhabited) | 42 |

B.3 Historical Fuel Accident at MNRC

The MNRC has experienced (in the late 1990s) a single fuel element leakage in its operational history. The incident occurred approximately 30 minutes after starting up the reactor for 1 MW operations. There was no drop reactor tank level associated with this incident making the severity of this accident somewhere between the MHA and a single fuel element 24 hours after shutting down. No one was inside the reactor room during the incident so no worker uptakes took place. The event was quickly detected by the reactor room and stack CAMs which acted to switch the ventilation path to “recirculation” mode before becoming saturated. The total offsite release of radioactivity was thought to be small but could not be determined with any degree of accuracy.

Of interest was the dose rate measured outside of the reactor room door just after the fuel leak was measure to be 40 mRem/hr and produced no significant (<1 mrem/hr) dose readings at the MNRC fence line.