



STATE OF RHODE ISLAND AND PROVIDENCE PLANTATIONS

RHODE ISLAND ATOMIC ENERGY COMMISSION

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Mr. William B. Kennedy, Project Manager
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Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D.C. 20555-0001

August 18, 2010

Re: Renewal of License No. R-95
Docket No. 50-193

Dear Mr. Kennedy:

We are responding to your requests for additional information (RAI) regarding our analysis of the maximum hypothetical accident (MHA). Since the requests basically questioned our analysis, we decided to update the analysis removing much of the conservatism in our original submittal. The attachment contains that updated analysis document and is intended to address RAI 13.2, RAI 13.3, RAI 13.4 and RAI 13.5.

If you have additional questions, please address them to the undersigned.

Very truly yours,

Henry J. Bicehouse, Assistant Director for Radiation and Reactor Safety
Rhode Island Atomic Energy Commission

I certify under penalty of perjury that the representations made above are true and correct.

Executed on: 8/18/10

By: Henry J. Bicehouse

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MRR

ATTACHMENT: FUEL DAMAGE RADIOLOGICAL ASSESSMENT

Introduction

The ultimate goal of any radiological assessment is to show the relationship between the source term, or quantity and types of released radionuclides and the dose equivalent resulting from that release. The assessment process must proceed in a logical fashion, following the radioactive material from its point of origin through its pathways to its resulting estimated total dose equivalent to both occupationally-exposed individuals and members of the public.

This assessment postulates an accident involving a fuel element at the Rhode Island Nuclear Science Center (RINSC) that releases fission products into the pool and subsequently out to the environment. The fuel element is postulated to have been damaged sufficiently to allow release of some of its accumulated fission products to the pool. Fission products include volatile elements such as iodine and various noble gases, water-soluble elements such as cesium and rubidium and insoluble elements such as many of the rare earth elements.

The scenario postulates a failure of the cladding. Cladding is the outer layer of the fuel rods, standing between the coolant and the nuclear fuel. In power reactors, it is usually made of a corrosion-resistant material with a low absorption cross section for thermal neutrons (zircaloy). However, the cladding for many research reactors (including the RINSC reactor fuel) is aluminum.

Regulatory Guidance

There has been a significant change of the US Nuclear Regulatory Commission's (NRC) regulatory position in using accident source term for radiological assessment following a design basis accident (DBA) since our Safeguards Report. On April 20, 1992, the NRC staff presented to the NRC commissioners the draft Revised Accident Source Terms for Light-Water Nuclear Power Plants. This effort is documented in SECY-92-127 and provided the first official position taken by the NRC in this matter since the publication of TID-14844.

TID-14844 was promulgated in 1962, and used for more than 30 years in evaluating the radiological consequences of a reactor accident. The TID-14844 model assumes that 100% of the noble gases and 50% of the iodine are instantaneously released to the containment and are available for leakage to the environment. TID-14844 was formally embodied in the NRC's regulations in parts 10 CFR 100 (siting) and 10 CFR 50 (review of control room habitability, post-accident shielding and sampling systems). It was also embodied in a host of NRC Regulatory Guides and NUREG reports that address off-site consequences of releases of radioactivity, equipment qualification, and other post-accident radiological concerns.

During the TMI-2 accident, fission products did not behave as predicted by the analytical methods and assumptions delineated in Regulatory Guides 1.3 and 1.4 and TID-14844. Although over 50% of the core was in a molten state, only about 55% of the highly volatile fission products and noble gases were released from the reactor vessel with a major portion retained in the reactor building. Less than 5% of the medium and low volatile fission products were released from the reactor vessel. It is now generally accepted that the chemical conditions in the reactor vessel were "reducing" in nature as opposed to "oxidizing." The elemental iodine was driven or converted to the iodide ion and readily combined with available metallic ions. The water soluble nature of the chemical forms prevented a major release of iodine to the atmosphere of the containment or auxiliary buildings and only a few curies were released to the environment.

To replace the instantaneous source term of TID-14844, the time-dependent source term of NUREG-1465 was introduced in 1995, which represents the accident source term enveloping all light water reactor plants. In the meantime, the radiological acceptance criteria for reactor site evaluation in 10 CFR Part 100 were also revised. In particular, the concept of a total effective dose equivalent (TEDE) has been incorporated in accordance with the radiation protection standards set forth in revised 10 CFR Part 20.

Subsequently, the publication of Regulatory Guide 1.183 and the revision of the Standard Review Plan 15.0.1 followed in 2000, which provided the licensee of an operating nuclear power reactor with the acceptable guidance for applying the revised source term. The guidance allowed the holder of an operating license issued prior to 10 January 1997 to voluntarily revise the accident source term used in the radiological consequence analyses of DBA.

Appendix B¹ of Regulatory Guide 1.183 provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a fuel handling accident at a light water reactor. The fuel handling accident in Appendix B postulates damage to fuel rods and is somewhat analogous to the postulated fuel damage accident being evaluated here. It should be noted that there are distinct differences between the types of fuel involved and the inventory of fission and activation products contained in the damaged fuel element.

Facility Description

The RINSC reactor uses a plate-type fuel design derived from the original Materials Test Reactor (MTR) in Idaho. The RINSC reactor is an open-pool design with fuel plates arranged in a rectangular array. Its design has the following characteristics:

1. The reactor has a prompt negative temperature coefficient and negative void coefficient.

¹ Appendix B, "Assumptions For Evaluating The Radiological Consequences Of A Fuel Handling Accident," NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors"

2. Only materials whose properties are well known to be stable under the operating conditions of the facility, including radiation exposure, were used for the fuel, coolant and safety-related structures.
3. Instrumentation and controls have been provided so that the operators know and have control over the status of the reactor at all times. Sufficient redundancy has been provided so that the loss of key instruments or controls will not deprive the operators of needed information or prevent shutdown of the reactor.
4. The facility was built and its equipment installed in a manner consistent with the highest standards of engineering practice.
5. Components were designed and installed in a way that permits continuous or periodic monitoring and/or inspection for signs of wear and incipient failure and which permits the periodic testing of components.

The reactor is a 2-MW_{th}, open pool, water-cooled (natural and forced convection mode) research reactor designed and built by General Electric. The fuel is low-enrichment uranium (LEU) and is reflected with a combination of graphite and beryllium. The fuel is U₃Si₂-Al enriched to less than 20% Uranium-235. The uranium silicide formulation retains most mixed fission products and retards the leakage of halogens and noble gas. The blister threshold temperature² measured on U₃Si₂ fuel plates ranges from 515-575 C (959-1067 F). This temperature range is well beyond anything encountered in operation.

The reactor core is located near the bottom of a 32-foot deep, aluminum-lined concrete pool. The reactor uses demineralized water as the primary coolant, shielding, and reactor moderator. The core is cooled by natural convection at power levels up to 100 kW, and by forced convection at power levels above 100 kW. Water levels are maintained a minimum of 23.54 feet above the active core during operation.

The structure surrounding the reactor constitutes a confinement building rather than providing absolute containment. Because of the low fission-product inventory, leakage from the structure can be tolerated.

Assumptions

We will assume that the reactor is operating normally under applicable technical specifications before the fuel is damaged. We will further assume that instruments, controls, and automatic protective systems are operating normally and available to help mitigate the consequences of the fuel damage.

Since radionuclides have physical and chemical properties identical to their stable elemental counterparts, we will assume that their transport will faithfully follow that of the stable element. From that point of view, the important parameters are the physical state (whether liquid, solid, or gaseous), the chemical form, solubility in air and water, oxidation states, sorption properties and volatility. In addition, engineered safety features provide additional barriers to the off-site release of fission products formed in the fuel

² NUREG-1313

element. Thus, there are several reductions in the source term available to mitigate the dose to the public from a release of fission products from a damaged fuel plate:

- The reactor fuel itself
- The pool water
- The confinement building
- The emergency exhaust system with its HEPA and charcoal filtration
- Atmospheric dilution caused by a 115-foot stack and its related atmospheric dispersion coefficient

In order to evaluate the consequences of fission product releases in the event of some damage to a fuel element occurring, we will assume that one plate in an element (from the aspect of vulnerability to mechanical damage, an outside plate would be most likely involved) is damaged to such an extent that total cladding integrity is lost and that volatile fission products are completely available for release to the primary coolant. (This situation is difficult to conceive because even if the cladding were completely ineffective, the metallic fuel matrix provides a substantial barrier against diffusion of fission products within the element.) For the plate type fuel utilized in the RINSC core, physical core damage to an element is possible via pitting of plate cladding or inadvertent mechanical shock; for instance, scratching of a plate surface in handling; such damage could lead to the escape of fission products. While the real magnitude of such damage is indeterminate, it is difficult to conceive of any situations or processes within the scope of reactor operations that could result in the release of a large fraction of the core fission product inventory.

It should be noted that as a holder of an operating license issued prior to January 10, 1997, we are allowed by 10 CFR 50.67, "Accident Source Term," to voluntarily revise the accident source term used in design basis radiological consequence analyses.

For purposes of dose equivalent estimation, we will assume that the reactor has operated sufficiently for the fission product inventory to reach saturation and that release of fission products in the damaged fuel element generally follows the empirical data generated by the University of Virginia during a fuel failure at their facility. No correction is applied for radioactive decay during transit. We will also assume that noble gases are unaffected by the pool. The pool is assumed to retain 99.5% of the iodine in terms of that which gets released from the pool to the confinement air and it is composed of 45% elemental and 55% organic species. All other fission and activation products are assumed to be retained in either the fuel or the pool.

Calculation of a Source Term

The radionuclide production and decay rate for the i^{th} radioactive isotope in the whole core is given by $\lambda_i N_i$ where:

$$\lambda_i N_i = (3.1 \times 10^{11} \text{ f sec}^{-1}) (2 \times 10^6 \text{ watts}) (\gamma_i)$$

and where:

$$\lambda_i = \text{radioactive decay constant for the } i^{\text{th}} \text{ radionuclide}$$

N_i = saturation number of atoms of the i^{th} radionuclide present

γ_i = fractional fission yield for the i^{th} radionuclide

f = fissions

But $\lambda_i N_i$ = activity in Becquerels since saturation was assumed. The following table summarizes the radionuclides present in a fuel plate that are both volatile and long enough lived to potentially escape from the fuel into the pool.

Nuclide	λ_i (/ sec)	γ_i (fission yield)	Core Activity (Becquerels)	Core Activity (Ci)	Single Plate (Ci)
I-131	9.73E-7	0.029	1.80E+16	4.86E+5	1.58E+3
I-132	8.02E-5	0.043	2.67E+16	7.22E+5	2.34E+3
I-133	9.25E-6	0.065	4.03E+16	1.09E+6	3.54E+3
I-134	2.20E-4	0.080	4.96E+16	1.34E+6	4.35E+3
I-135	2.88E-5	0.064	3.97E+16	1.07E+6	3.47E+3
Kr-85m	4.41E-5	0.013	8.06E+15	2.18E+5	7.08E+2
Kr-85	2.05 E-9	0.000255	1.58E+14	4.27E+3	1.39E+1
Kr-87	1.48E-4	0.025	1.55E+16	4.19E+5	1.36E+3
Kr-88	6.95E-5	0.036	2.23E+16	6.02E+5	1.95E+3
Xe-131m	6.67E-7	0.029	1.80E+16	4.86E+5	1.58E+3
Xe-133m	3.50E-6	0.065	4.03E+16	1.09E+6	3.54E+3
Xe-133	1.53E-6	0.065	4.03E+16	1.09E+6	3.54E+3
Xe-135m	7.40E-4	0.064	3.97E+16	1.07E+6	3.47E+3
Xe-135	2.11E-5	0.064	3.97E+16	1.07E+6	3.47E+3

Pathway Overview

The basic pathway for release of radionuclides to the environment is fuel element to pool water to confinement air to the environs following passage through high efficiency particulate filters and a charcoal filter and atmospheric dispersion from a 115-foot stack.

Applicable Empirical Data

The University of Virginia operated a research reactor of a virtually identical design for many years. Experience at the University of Virginia with an earlier generation fuel design supports the conclusion that a release of a large fraction of the core fission product inventory is unlikely.

In a fission gas release from a fission plate at the University of Virginia reactor on May 3, 1968, isotopic concentrations of noble gases were measured in the reactor room, amounting to ~0.1% of the total fission plate inventory. Iodine isotope concentrations were too low to be detected. However, the University of Virginia concluded from

instrument readings that the iodine isotopes were released from the pool in amounts less than 10% of the concentrations of noble gases.³

If we use this empirical data from that earlier generation of a similar fuel element design, we would estimate the release of the noble gases to be 1 E-3 times the noble gas activity in the plate. For example, the Krypton-85m activity released would be 708 curies times 1 E-3 or approximately 708 millicuries. Iodine activity released would be 1 E-4 times the iodine activity in the plate also based on the University of Virginia's experience. The following table summarizes the activities released from the pool using this empirical data.

Nuclide	Single Plate (Ci)	UVA Release Fraction	Release Using UVA Empirical Data (Ci)	Release (μ Ci)
I-131	1.58E+3	1 E-4	1.58E-1	1.58E+5
I-132	2.34E+3	1 E-4	2.34E-1	2.34E+5
I-133	3.54E+3	1 E-4	3.54E-1	3.54E+5
I-134	4.35E+3	1 E-4	4.35E-1	4.35E+5
I-135	3.47E+3	1 E-4	3.47E-1	3.47E+5
Kr-85	1.39E+1	1 E-3	1.39 E-2	5.14 E+4
Kr-85m	7.08E+2	1 E-3	7.08E-1	7.08E+5
Kr-87	1.36E+3	1 E-3	1.36	1.36E+6
Kr-88	1.95E+3	1 E-3	1.95	1.95E+6
Xe-131m	1.58E+3	1 E-3	1.58	1.58E+6
Xe-133m	3.54E+3	1 E-3	3.54	3.54E+6
Xe-133	3.54E+3	1 E-3	3.54	3.54E+6
Xe-135m	3.47E+3	1 E-3	3.47	3.47E+6
Xe-135	3.47E+3	1 E-3	3.47	3.47E+6

Reactor Pool

The reactor pool is 32 feet deep, 22½ feet long and 8½ feet wide with a capacity of 36,000 gallons and an aluminum liner. In operation, it contains approximately 33,000 gallons of demineralized water used as primary coolant, shielding, and reactor moderator.

The two primary noble gas fission products (krypton and xenon) and the halogens (primarily iodine) are somewhat soluble in demineralized water. Krypton has a solubility of 59.4 cubic centimeters per kilogram of water.⁴ Xenon has a solubility of 108.1 cubic centimeters per kilogram of water.⁵ Iodine is minimally soluble in water (0.03 g/100 g water at 20°C) and does not react with it.⁶

The mass of any radionuclide can be calculated:

³ University of Virginia Reactor Final Safety Analysis Report, pg. 83 (1970)

⁴ Greenwood, Norman N. and Earnshaw, A. *Chemistry of the Elements* (2nd edition) 1997

⁵ Ibid

⁶ Ibid

$M = \frac{QW}{\lambda N}$ where Q is the activity in becquerels; W is the radionuclide's atomic weight; λ is the radionuclide's decay constant; and N is Avogadro's Number (6.02E23 atoms/gram/atom). Thus, it should be noted that the masses of the iodine and noble gas inventories in any single plate total only a few milligrams and the pool contains 33,000 gallons of water. It should also be noted that noble gas behavior in water generally follows Henry's law⁷.

The gaseous phase of a radionuclide tends to escape to the atmosphere via the air-water interface. If the radionuclide is uniformly mixed over the water column H, this escape mechanism can be represented by a single term:

$\left(\frac{dC}{dt}\right) = K(C - C_s)$ where C is the vertically uniform radionuclide gas concentration and C_s is its saturation value. C_s is governed by Henry's law in equilibrium with the partial pressure of that gas in the surrounding atmosphere (usually, $C_s = 0$ for radionuclides). K is a depth average loss coefficient that is related to the actual surface transfer coefficient K_L by $K = K_L/H$. Useful estimates for K can be obtained from oxygen exchange although there is some uncertainty as to how molecular differences between diatomic oxygen molecules (O_2) and monatomic noble gas molecules or diatomic iodine molecules affect the transfer process. K_L for oxygen has been measured at 9-10 mg/cm^3 . H is approximately 717 cm in the pool assuming the postulated cladding failure occurs at the very top of the active core. Thus, the loss coefficient (K) is 10 mg/cm^3 divided by 717 cm or about 0.014 mg/cm^2 .

Thus, we have a very small amount of solute in a very large amount of solvent. By that reasoning, we should see retention of both the iodine and noble gas molecules by the pool delaying release and decreasing the source term. Delay allows the shorter half-life radionuclides to decay and thus reduces the available source term. This is consistent with the empirical data from the University of Virginia and the reported data from the NRC Regulatory Guide 1.183.

Confinement Building

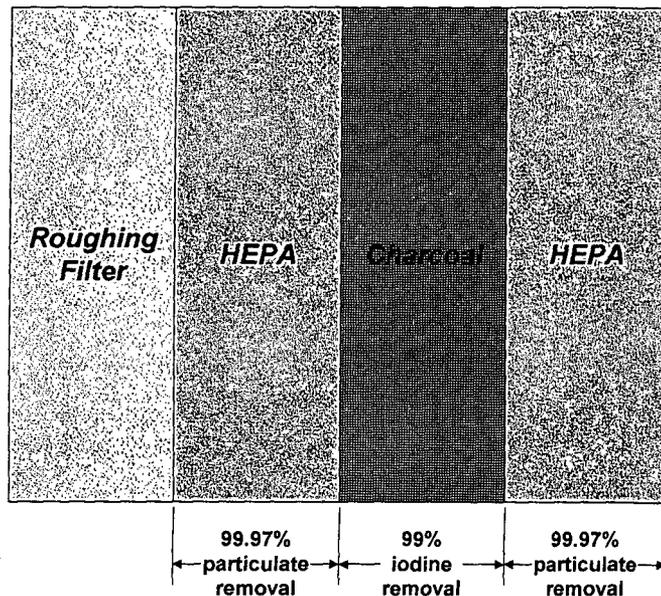
Normal operation isolates the reactor building and maintains a negative pressure within. When an unsafe radiological situation develops as defined in the facility operating and emergency procedures, the confinement system is activated by depressing any one of five emergency evacuation buttons. The evacuation alarm horns are sounded and personnel respond in accordance with the operating and emergency procedures. In the unlikely event of a release of fission products, or other airborne radioactivity, the confinement isolation initiation system secures the normal ventilation exhaust fan, bypasses the normal ventilation supply up the stack, and closes the normal inlet and exhaust valves. In confinement, the emergency exhaust system maintains a negative building pressure with a combination of controls intended to prevent unloading any large fraction of airborne activity.

⁷ Henry's Law states that at a constant temperature, the amount of a given gas dissolved in a given type and volume of liquid is directly proportional to the partial pressure of that gas in equilibrium with that liquid.

Treatment

The emergency exhaust purges the building air through charcoal and absolute filters and controls the discharge by diluting with supply and through a 115-foot stack. The absolute filters remove 99.97% of the particulate activity while the charcoal filter removes over 99% of the iodine. Un-decayed noble gases are unaffected by the absolute filters but are slowed by the charcoal filter.

Reactor Building Exhaust Cleanup System



When activated, the reactor building exhaust cleanup system exhausts air from the reactor building through a roughing filter, a HEPA filter, a charcoal filter for removing iodine, and a second HEPA filter to remove any entrained charcoal dust contaminated with radioactive iodine. Each HEPA filter is tested to remove 99.97% of the particulates. The charcoal filter removes 99% of the entrained radioactive iodine. The exhaust is discharged through a 115-foot high stack.

When the reactor building cleanup system is in operation, the exhaust air discharged to the outside should contain no more than 2.5×10^{-7} times the concentration of airborne particulates present in the reactor building air. The exhaust air should contain no more than 1×10^{-2} times the iodine concentration present in the reactor building air.

Atmospheric Dispersion

Leakage air from the building is assumed to be dispersed in the atmosphere according to quite restrictive meteorological considerations. The release is assumed to be at ground level and dilution is assumed to occur under Pasquill Type F condition and a wind speed of one meter per second. Wind direction is assumed to be constant over the entire duration of the release. The calculated value of the concentration reduction factor, X/Q , has been divided by three to yield an effective, concentration reduction factor, $(X/Q)_{\text{eff}}$, which makes some allowance for building dispersal effects. The minimum radius defining the area under the immediate control of RINSC is 48 meters and this is the distance at which $(X/Q)_{\text{eff}}$ has been calculated. X/Q is calculated according to

$$X/Q = \frac{1}{\Pi \sigma_y \sigma_z u}$$

Values of $\sigma_y = 2.0$ m and $\sigma_z = 1.2$ m for 48 meters have been estimated by extrapolation of the curves given in U.S. Atomic Energy Commission's "Meteorology and Atomic Energy 1968"⁽²⁾, pp. 102-103 for Pasquill Type F condition:

$$x = \frac{1}{\Pi(2.0\text{m})(1.2\text{m})(1\text{m sec}^{-1})} = 0.133 \text{ sec m}^{-3} \text{ or } 1.33 \times 10^{-7} \text{ sec cm}^{-3}$$

$$\text{and } (x) = X/3Q = 4.4 \times 10^{-8} \text{ sec cm}^{-3}$$

Total Effective Dose Equivalents to Staff Members

Personnel in the Containment Building would pass through the airborne noble gases and halogens during their evacuation.

Nuclide	Release (μCi)	Reactor Bldg. Conc. ($\mu\text{Ci}/\text{cm}^3$)	Occ. DAC ($\mu\text{Ci}/\text{cm}^3$)	Dose Rate (DAC- Hrs/Hr)	Evacuation Dose (DAC- Hours)
I-131	1.58E+5	2.57E-5	3E-7	85.67	7.14
I-132	2.34E+5	3.80E-5	3E-6	12.67	1.06
I-133	3.54E+5	5.76E-5	1E-7	576	48
I-134	4.35E+5	7.10E-5	2E-5	3.55	0.30
I-135	3.47E+5	5.64E-5	7E-7	80.57	6.71
Kr-85	5.14E+4	8.36E-5	1E-4	0.84	0.07
Kr-85m	7.08E+5	1.15E-4	2E-5	5.75	0.48
Kr-87	1.36E+6	2.21E-4	5E-6	44.2	3.68
Kr-88	1.95E+6	3.17E-4	2E-6	158.5	13.21
Xe- 131m	1.58E+6	2.57E-4	4E-4	0.64	0.05
Xe- 133m	3.54E+6	5.76E-4	1E-4	5.76	0.48
Xe-133	3.54E+6	5.76E-4	1E-4	5.76	0.48
Xe- 135m	3.47E+6	5.64E-4	9E-6	62.67	5.22
Xe-135	3.47E+6	5.64E-4	1E-5	56.40	4.70

Noble gases are an immersion hazard. The derived air concentration (DAC) for any noble gas is based upon an infinite cloud producing 2.5 mrem/DAC-hour. Evacuation drills have shown that the Containment Building can be evacuated in less than 5 minutes. Based on the activities in Containment shown above, the dose equivalent rate to a staff member inside Containment would be 27.89 DAC-hours/hour times 2.5 mrem per hour per DAC-hour. Evacuation results in a deep dose equivalent of 69.7 mrem.

Radioiodine is an inhalation hazard. The committed dose equivalent to the thyroid can be estimated by assuming that the thyroid would receive 50 rems were the individual to

breathe air at the derived air concentration for one year. Thus, one DAC-hour of any radioiodine would result in committed dose of 25 mrem for an occupational exposure. The total committed dose equivalent to the thyroid from the radioiodine would be 56.21 DAC-hours/hour times 25 mrem/DAC-hour or 1,405 mrem.

The committed effective dose equivalent from the radioiodine must be calculated individually.

Radionuclide	Concentration In Confinement Air ($\mu\text{Ci}/\text{cm}^3$)	Volume of Air Inhaled (cm^3) ⁸	Intake (μCi) ⁹	Committed Effective Dose Equivalent mrem per μCi ¹⁰	CEDE (mrem) ¹¹
I-131	2.57E-5	1 E+4	0.26	33	8.58
I-132	3.80E-5	1 E+4	0.38	0.011	0.004
I-133	5.76E-5	1 E+4	0.58	5.85	3.39
I-134	7.10E-5	1 E+4	0.71	0.13	0.09
I-135	5.64E-5	1 E+4	0.56	1.23	0.69

The Committed Effective Dose Equivalent is simply the sum of the individual effective dose equivalents from the radioiodine or 12.75 mrem. The total effective dose equivalent is equal to the deep dose equivalent plus the committed effective dose equivalent or 82.47 mrem. Since any individual working in the Containment Building is considered occupationally exposed, the dose equivalents meet 10 CFR 20 requirements.

Offsite Doses

Nuclide	Reactor Bldg. Conc. ($\mu\text{Ci}/\text{cm}^3$)	Air Conc. Released from the Stack ($\mu\text{Ci}/\text{cm}^3$) ¹²	Release Rate ($\mu\text{Ci}/\text{sec}$) ¹³	Concentration at 48 meters ($\mu\text{Ci}/\text{cm}^3$) ¹⁴
I-131	2.57E-5	2.57E-7	0.46	4E-9
I-132	3.80E-5	3.80E-7	0.68	3E-9
I-133	5.76E-5	5.76E-7	1.04	5E-9
I-134	7.10E-5	7.10E-7	1.28	6E-9
I-135	5.64E-5	5.64E-7	1.02	5E-9
Kr-85	8.36E-5	8.36E-5	150	6E-6
Kr-85m	1.15E-4	1.15E-4	207	9E-6
Kr-87	2.21E-4	2.21E-4	398	2E-5

⁸ 2,000 ml/min x 5 min = 10,000 ml

⁹ Concentration in Confinement Air x Volume of Air Inhaled

¹⁰ Values taken from Table 2.1, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors For Inhalation, Submersion, and Ingestion*, EPA-520/1-88-020 (September 1988)

¹¹ CEDE = Intake in μCi x Committed Effective Dose Equivalent per μCi

¹² Noble Gases are unaffected by treatment system; Iodine concentrations are 1%

¹³ The emergency exhaust rate is typically 4,000 cfm or 1.8 E+6 cm^3/sec

¹⁴ Concentration at 48 meters = Release Rate x X/Q at 48 meters = Release Rate in $\mu\text{Ci}/\text{sec}$ x 4.4 E-8 sec/cm^3

Kr-88	3.17E-4	3.17E-4	571	3E-5
Xe-131m	2.57E-4	2.57E-4	463	2E-5
Xe-133m	5.76E-4	5.76E-4	1037	5E-5
Xe-133	5.76E-4	5.76E-4	1037	5E-5
Xe-135m	5.64E-4	5.64E-4	1015	4E-5
Xe-135	5.64E-4	5.64E-4	1015	4E-5

The following table shows the thyroid and committed effective dose equivalents if a member of the general public remains for two hours at the boundary. The doses are well within allowable doses to the general public in 10 CFR 20.

Nuclide	Concentration at 48 meters ($\mu\text{Ci}/\text{cm}^3$)	Intake in 2 hours (μCi) ¹⁵	Thyroid ¹⁶ (mrem/ μCi)	Resulting Dose to Thyroid (mrem)	CEDE (mrem/ μCi)	Committed Effective Dose Equivalent ¹⁷ (mrem)
I-131	4E-9	0.00096	1080	1.03	32.89	0.032
I-132	3E-9	0.00072	6.43	0.005	0.38	0.00027
I-133	5E-9	0.0012	180	0.22	5.85	0.0070
I-134	6E-9	0.0014	1.06	0.0015	0.13	0.00018
I-135	5E-9	0.0012	31.3	0.038	1.23	0.0015

In the following table, gamma ray doses are calculated on the basis of the submersion dose principle where the concentration of radioactivity at the 48 meter location is compared to the occupational DAC for each radionuclide of interest.

Nuclide	Concentration at 48 meters ($\mu\text{Ci}/\text{cm}^3$)	DAC ($\mu\text{Ci}/\text{cm}^3$)	DAC-hrs ¹⁸	Dose (mrem) ¹⁹
Kr-85	6E-6	1E-4	0.12	0.30
Kr-85m	9E-6	2E-5	0.10	0.25
Kr-87	2E-5	5E-6	8.00	20.00
Kr-88	3E-5	2E-6	30.00	75.00
Xe-131m	2E-5	4E-4	0.10	0.25
Xe-133m	5E-5	1E-4	1.00	2.50
Xe-133	5E-5	1E-4	1.00	2.50
Xe-135m	4E-5	9E-6	8.88	22.20
Xe-135	4E-5	1E-5	8.00	20.00

¹⁵ Intake = Breathing Rate x Concentration at 48 meters; Breathing Rate of 2,000 cm^3/min x 120 minutes or 240,000 cm^3

¹⁶ Values taken from Table 2.1, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors For Inhalation, Submersion, and Ingestion*, EPA-520/1-88-020 (September 1988)

¹⁷ Ibid

¹⁸ Concentration at 48 meters divided by the DAC times two hours

¹⁹ DAC-hours times 2.5 mrem/DAC-hour

The total gamma dose to a member of the public who remains at the 48 meter perimeter for two hours is approximately 143 mrem. This dose is greater than the annual limit of 100 mrem but less than the 500 mrem the NRC allows a licensee to request for routine operations. Because of the assumptions concerning no decay in transit, major contributors to this dose include Krypton-87, Krypton-88, and Xenon-135m. Krypton-87 has a physical half-life of 76.3 minutes. Krypton-88 has a physical half-life of 2.84 hours. Xenon-135m has a physical half-life of 15.36 minutes. Thus, any delay allowing decay in transit would reduce the dose contributions from those radionuclides significantly. Since a fuel failure is not a routine operation, and the assumption of no decay results in a conservatively high dose estimate, the dose generally meets NRC requirements.

Summary & Conclusions

Research reactors are generally held to the dose limits in 10 CFR 20. This assessment has shown that the Rhode Island Nuclear Science Center's reactor meets those standards for a hypothetical fuel failure accident. With High Enriched Uranium fuel, the content of Uranium-238 is small and the production of plutonium and other actinides is negligible. The Low Enriched Uranium fuel has a content of more than 80% Uranium-238 and a greater potential for the buildup of plutonium with burn-up. Studies²⁰ have shown that though there may be a slight increase in dose due to plutonium buildup, the consequences are insignificant.

²⁰ W.L. Woodruff, D. K. Watinner, and J. E. Matos, "A Radiological Consequence Analysis with HEU and LEU Fuels"

Appendix 1: Regulatory Standards

The basic regulatory standards are found in 10 CFR 20.

§20.1201 Occupational dose limits for adults.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under §20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of --

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin, and to the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rems (0.50 Sv) to the skin or to any extremity.

(b) Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits for planned special exposures that the individual may receive during the current year (see §20.1206(e)(1)) and during the individual's lifetime (see §20.1206(e)(2)).

(c) The assigned deep-dose equivalent and shallow-dose equivalent must be for the part of the body receiving the highest exposure. The deep-dose equivalent, lens dose equivalent, and shallow-dose equivalent may be assessed from surveys or other radiation measurements for the purpose of demonstrating compliance with the occupational dose limits, if the individual monitoring device was not in the region of highest potential exposure, or the results of individual monitoring are unavailable.

(d) Derived air concentration (DAC) and annual limit on intake (ALI) values are presented in table 1 of appendix B to part 20 and may be used to determine the individual's dose (see §20.2106) and to demonstrate compliance with the occupational dose limits.

(e) In addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity (see footnote 3 of appendix B to part 20).

(f) The licensee shall reduce the dose that an individual may be allowed to receive in the current year by the amount of occupational dose received while employed by any other person (see §20.2104(e)).

§20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that--

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 millisievert) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released in accordance with §35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with §20.2003, and

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with §35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

(b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

(c) A licensee or license applicant may apply for prior NRC authorization to operate up to an annual dose limit for an individual member of the public of 0.5 rem (5 mSv). The licensee or license applicant shall include the following information in this application:

(1) Demonstration of the need for and the expected duration of operations in excess of the limit in paragraph (a) of this section;

(2) The licensee's program to assess and control dose within the 0.5 rem (5 mSv) annual limit; and

(3) The procedures to be followed to maintain the dose as low as is reasonably achievable.

(d) In addition to the requirements of this part, a licensee subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR Part 190 shall comply with those standards.

(e) The Commission may impose additional restrictions on radiation levels in unrestricted areas and on the total quantity of radionuclides that a licensee may release in effluents in order to restrict the collective dose.