PURDUE UNIVERSITY

SCHOOL OF NUCLEAR ENGINEERING

10 April 2013

Document Control Desk
US Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Attn: Ms. Cindy Montgomery, Research & Test Reactors (NRR/DPR/PRLB), Mailstop O12 D20

SUBJECT: PURDUE UNIVERSITY - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL (TAC NO. ME 1594), RESPONSES TO RAIs (ML103400115 and ML103400250)

Dear Ms. Montgomery:

Enclosed please find the responses to the Request for Additional Information regarding the Purdue University Reactor License Renewal dated 6 July 2011. Should you have any questions or require further information, please don't hesitate to call me at 765.496.3573, or e-mail at jere@purdue.edu.

I hereby certify under penalty of perjury with my signature below that the information contained in this submission is true and correct to the best of my knowledge.

Very respectfully,

/SA

Jere H. Jenkins
Director of Radiation Laboratories

Attachments: As described.

Cc: Duane Hardesty, USNRC Project Manager for PUR-1
Leah Jamieson; Purdue University College of Engineering
Jim Schweitzer, Purdue University REM, CORO Chair
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REQUESTED ADDITIONAL INFORMATION IN RESPONSE TO RAIs

REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL (TAC NO. ME 1594)

54 NUREG-1537, Part 1, Section 10 provides guidance for providing information on the administrative procedures used by the applicant to approve an experiment. These procedures should be discussed in detail in Chapter 10 of the SAR, summarized in Chapter 12, "Conduct of Operations," and included in the technical specifications. Please provide the experiment review and approval methodology and discuss the experiment review and approval process.

Response:

The present review and approval process as described in the PUR-1 operations manual is as follows:

> A "Request for Reactor Operation" form will be prepared and submitted before any reactor operation is performed. This form, properly filled in, will state the purpose, procedure, apparatus, intended power level, reactor conditions, and expected results of the experiment, with supporting reasons. The Reactor Supervisor will review the request and consult members of the scientific staff if needed to establish the type of experiment. The Reactor Supervisor will decide upon the safety of a proposed experiment unless review by the Reactor Operations Committee is requested. Upon satisfactory completion of review of a proposed experiment, the Reactor Supervisor will schedule a time for its performance. Each "Request for Operation" form will be signed by the experimenter and, when required, countersigned by the staff member advisor. After the form is reviewed and approved, it is checked for necessary signatures, and posted in the control room.

Since procedures are subject to changes with appropriate review and approval, we do not feel that discussion in detail of procedures is appropriate in the SAR; only an overview should be provided, otherwise a procedure change would require a change in the SAR with each revision. New experiments are reviewed by reactor staff to ensure the safety of the reactor, staff and experimenters, and the public and environment. Experiments are also reviewed against the PUR-1 technical specifications to ensure operations within appropriate limits.

69. The requirements of 10 CFR 20.1201 include limiting the total dose equivalent to facility staff and the public from licensed reactor operations. In Section 5.6 of the SAR, it states that no nitrogen-16 activity has been observed to date in the reactor room. This referenced observation is known to be at a power level of 1 kW, based on previous licensed power for PUR-1. Please provide an updated evaluation of a bounding safety analysis that explains all analyses, assumptions, and conclusions at the requested licensed power level for the maximum potential release of N-16 from the pool water into the reactor room and any potential dose to the facility staff and members of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.).

Response:

There is a negligible fast neutron flux in PUR-1, which is required for the production of N-16 via the $^{16}O(n,p)^{16}N$ reaction, even at the new requested power. However, in the unlikely event that N-16 is produced, using a NATCON analysis at 18 kW power (which is higher than the 12 kW requested licensed power level, but is assumed to be an enveloping calculation), the maximum flow rate at the outlet of the hot channel is 0.00686 kg/s, at a velocity of 19.2 mm/s. Assuming an extremely conservative straight-line

path of travel of a unit-volume of coolant water containing N-16, it would take approximately 206 seconds for that unit-volume to reach the surface of the pool, or approximately 28 half-lives for the produced N-16 (7.13 s). Thus, any credible assumed quantity of N-16 produced will have long since decayed before reaching the surface of the pool.

77. NUREG 1537, Part 2, Chapter 13 states credible accidents should be categorized and the most limiting accident in each group should be analyzed in detail including the potential consequences of the various accident scenarios including loss-of-coolant accident (LOCA) events.

A. Please provide an evaluation of a safety analysis of the LOCA accident sequence assuming the maximum licensed power level including uncertainty resulting from power level measurement uncertainty.

B. Please provide an evaluation of a safety analysis for safe cooling of the fuel during complete loss of coolant event at the peak fuel power densities for the maximum requested licensed power level.

C. Please provide an evaluation of a safety analysis for the slow draining process, which may result in a partially uncovered core (partial LOCA), that may not be cooled by assuming a continuous circulation of air. Please discuss a partial LOCA scenario and indicate whether the fuel temperature in a partially uncovered core is still bounded by the SAR LOCA analysis.

Response:

We feel that this question is unreasonable. As written in Section 13.1.3 of NUREG 1537, "In many non-power reactor designs, the loss-of-coolant accident (LOCA) is of no consequence because decay heat in the fuel is so small as to be incapable of causing fuel failure." NUREG 1537 goes on to describe that in some higher power reactors (normally greater than 2 MW), some engineered safety feathers for emergency core cooling may be necessary. The requested power uprate to 12 kW is 166 times smaller than the 2 MW threshold suggested by NUREG 1537 where fuel damage as a result of a LOCA is possible.

The reactor pool is designed to prevent unintentional drainage. The pool is constructed of a stainless steel liner and set in a second steel tank with the interstitial region filled with sand. The tank rests on a concrete pad about 4.6 m below the floor of the reactor room, which is in the basement of the building. The pool has no drains or coolant pipes below floor level (more than 8 feet above the core) that could open or break. Therefore, a sudden loss of coolant is considered to be extremely unlikely. Furthermore, if the pool drained instantaneously while the reactor was operating, the loss of water (moderator) would shut down the reactor.

Even if the worst case is assumed, and PUR-1 experiences a LOCA, utilizing the Way-Wigner [1] equation for fractional power resulting from core decay heat:

$$\frac{P}{P_o} = 6.22\times10^{-2}\left[t^{-0.2} - (T_i + t)^{-0.2}\right] \quad (1)$$

where

P	=	Core power after shutdown
P_o	=	Power generated during operation
t	=	Time in seconds after shutdown, and
T_i	=	Time irradiated, or time at operating power.

Assuming an infinite time at a conservative operating power of 18 kW (50% above the requested 12 kW), the power generated in the reactor at 60 seconds after shutdown is 493 W, or an average of 2.6 W/plate for the 190 plates. Even using a conservative power peaking factor of 2 for the hot channel plate, applying that value of 5.2 W/plate to all plates in the reactor, and assuming adiabatic conditions (which would encompass all conceivable LOCA scenarios), there would not be significant enough heating to cause damage to the fuel plates in any credible scenario.

78. NUREG-1537, Part 1, Section 13.1.4 provides guidance for analysis of loss-of-coolant flow resulting from blocked fuel cooling channels.

A. Please provide an evaluation of a safety analysis that provides a complete assessment of the potential for fuel channel blockages and how adequate heat transfer during such blockages is maintained.

B. Please discuss facility procedures or any other blockage-mitigating PUR-1 design features for foreign material exclusion from entry to the reactor pool in order to prevent blockage of coolant channels.

Response:

NUREG 1537, Part 1, Section 13.1.4 provides guidance for analysis of loss-of-coolant flow as "most limiting for forced-convection non-power reactors, where the forced flow is downward through the reactor core." Since PUR-1 operates with only natural convection, there is no scenario to be considered for loss of forced flow. Using a NATCON analysis at 18 kW power (which is higher than the 12 kW requested licensed power level, but is assumed to be an enveloping calculation), the maximum flow rate at the inlet of the hot channel is 0.00686 kg/s, at a inlet velocity of 19.13 mm/s. In order for a channel to be blocked at the inlet, a buoyant item would have to find its way under the reactor deck fifteen feet below the surface of the pool. This is not a credible scenario. It is also not a credible scenario for any non-buoyant item that might find its way to the bottom of the pool to be drawn up from the bottom of the pool to block a channel due to the mass flow rate. Therefore, a loss-of-coolant flow accident is not a necessary consideration.

92. SAR, Section 13.2.1, makes reference to restricted and unrestricted areas. These types of areas are not defined in the SAR or emergency plan. Please update the SAR and/or emergency plan to use consistent designations or provide the definition of these areas and explanation of relationship to defined areas such as the operations boundary, site boundary, reactor building, or nuclear engineering lab.

Response:

There are no suggested definitions of 'restricted area' or 'unrestricted area' in NUREG 1537 or the ANSI/ANS 15.1, 15.16 or 15.21 standards, nor is there guidance that suggests they be provided. The use of 'restricted area' and 'unrestricted area' in the SAR, Section 13.2.1, is consistent with the Accident Analysis guidance in Chapter 13 of NUREG 1537, and is in accordance with 10 CFR 20. The definitions of 'restricted area' and 'unrestricted area' as presented in 10 CFR 20.1003 are as follows:

> *Restricted area* means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Restricted area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.

> *Unrestricted area* means an area, access to which is neither limited nor controlled by the licensee.

It should be assumed that a 'restricted area' is wherever it needs to be defined as determined by reactor staff (as suggested in the Emergency Plan) where exposures to personnel or the public are possible as a result of an accident.

References

1. Etherington, H., *Nuclear engineering handbook*. 1st ed. McGraw-Hill handbooks. 1958, New York,: McGraw-Hill. 1 v. (various pagings).

REQUESTED ADDITIONAL INFORMATION IN RESPONSE TO RAIs

REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL (TAC NO. ME 1594)

71. The requirements of 10 CFR 20.1101 states that each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities, in order to limit the total effective dose equivalent to facility workers (annual occupational dose less than 5 rem) and the total effective dose equivalent to individual members of the public (annual public dose less than 100 mrem). Please provide an evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential estimate of the total annual production of argon-41 from PUR-1 normal operations. In addition, please evaluate and discuss the potential maximum dose to a facility worker and to a member of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.) due to this bounding yearly production and release of argon-41 from the facility.

Response:

A revised Chapter 11 is attached to this document.

74. Section 13.3 of NUREG-1537 states "Maximum acceptable non-conservative instrument error may be assumed to exist at accident initiation." Please provide all accident analysis calculation results performed with the assumption that the power level is the maximum requested power level plus the justified uncertainty resulting from power level measurement and other uncertainty.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

75. NUREG-1537, Section 13 provides guidance to the licensee to identify the limiting event for each accident group and to perform quantitative analysis for that event. Please identify and discuss the potential methods whereby excess reactivity could be inserted into the reactor to cause an excursion. Please provide an evaluation of a safety analysis that quantitatively evaluates the most limiting event for addition of excess reactivity.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

76. NUREG-1537, Section 13.1.2 provides guidance for analysis of insertion of excess reactivity. In Section 13.2.2 of the SAR the licensee reports two reactivity insertion scenarios: the first with scram, the second without scram.

A. For the scram assumption scenario, the accident is assumed by PUR-1 to be initiated at a power of 10 kW and scram is assumed to occur on a power trip at 12 kW. This assumption, however, does not appear to account for the power measurement error and uncertainty.

Please revise the assumption for this calculation and provide an evaluation of a safety analysis at the requested maximum licensed power level including power level measurement uncertainty.

B. For the no-scram assumption scenario, the initial power assumed by PUR-1 before the accident is 10 kW and does not appear to account for the power measurement error and uncertainty. Please revise the assumption for this calculation and provide an evaluation of a safety analysis at the requested maximum licensed power level including power level measurement uncertainty.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

80. NUREG-1537, Section 13 provides guidance to the licensee to consider accident initiators that do not fall into the other categories, such as operator errors, instrument or control malfunctions, electrical faults, and others. Please provide an evaluation of a safety analysis using the guidance of NUREG-1537, Section 13.1.7 that considers the range of PUR-1 operations for possible scenarios involving the potential for equipment malfunction and potential consequences.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

83. NUREG-1537, Part 1, Section 13.1.5 provides guidance for analyzing accidents for mishandling or malfunction of fuel for non-power reactors. Please provide an evaluation of a safety analysis that provides the MHA fission product inventories at the requested maximum PUR-1 licensed power, including power level measurement uncertainty.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

85. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. Please explain the use of the fission product inventory from Table A-1 in "Nuclear Power Reactor Safety" (E.E. Lewis, 1977) as the basis instead of calculating the inventory in the experiment or the fuel. Please provide an evaluation of a safety analysis that substantiates PUR-1's chosen method of establishing the fission product inventory for the singly encapsulated experiment or the fuel is reliable and conservative.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

86. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The NRC staff is unable to reproduce the dose rate results for a failed fueled experiment to the facility workers due to a 1.5 minute exposure in the reactor room for the thyroid dose, the skin dose, and the whole body dose of Table 13-2 in the SAR. Please provide more details for these calculations including verifying that these results are applicable to the maximum requested licensed power for PUR-1 operations.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

87. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The NRC is unable to reproduce the ©, ®and thyroid dose rates based on the dispersion factor of 1.78x10-4 s/m3 tabulated in Table 13-3 of the SAR. Please provide more details for these calculations including verifying that these results are applicable to the maximum requested licensed power for PUR-1 operations

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

88. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The NRC is unable to reproduce the results for a dispersion factor (DF) of 1.78x10-4 s/m3, using Pasquill Type F conditions, wind speed of 1 m/s, and a downstream receptor distance of x = 100 m provided in Section 13.2.1, p.13-9 of the SAR. Please provide more details for the derivation and calculation to obtain the stated result in the SAR.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

89. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1 Maximum Hypothetical Accident (Failure of a Fueled Experiment) states on page 13-7 "The calculated saturation activity for each respective radioisotope and its concentration is the Reactor Room after experiment failure is shown in Table 7.5." Table 7.5 does not exist in SAR. Please update the SAR to provide the referenced saturation activities or state where the information exists in the SAR.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

90. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states "The radioactive material content, including fission products, of any singly encapsulated experiment should be limited…"

Please provide an evaluation of a safety analysis that demonstrates the radioactive material content will be limited to meet annual doses stated in 10 CFR Part 20 and will not be exceeded for release of all gaseous, particulate, or volatile components from any singly encapsulated experiment.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

91. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states "The radioactive material content, including fission products, of any doubly encapsulated experiment or vented experiment should be limited…". Please provide an evaluation of a safety analysis that demonstrates the radioactive material content will be limited to justify annual doses stated in 10 CFR Part 20 will not be exceeded for release of all gaseous, particulate, or volatile components from any doubly encapsulated experiment.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

93. NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states the dose to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release. Please provide an evaluation of a safety analysis that substantiates these stated doses for members of the public who may be in the uncontrolled areas of the Engineering building during an MHA event.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

94. SAR, Section 13.2.1 discusses doses to any person occupying an unrestricted area continuously for a period of two hours. Please provide an evaluation of a safety analysis that justifies the basis for the 2 hour limit and discuss any requirements for evacuation of the unrestricted area, if applicable.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. SAR, Section 13.2.1, page 13-8 states the dose to any person occupying a "restricted area" during the length of time required to evacuate. Please provide an evaluation of a safety analysis that substantiates these stated doses for personnel, including members

of the public who may be in the restricted areas of the Engineering building during an MHA event.

Response:

Chapter 13 has been rewritten, and is included as an attachment in this document.

11 RADIATION PROTECTION AND WASTE MANAGEMENT

11.1 Radiation Protection Program

Purdue University has a structured radiation safety program. Policies for the program are determined by the University Radiation Safety Committee, which has the mission to ensure the safety of the University and community in the utilization of all radioactive materials and radiation producing devices at the University by faculty, staff or students. This includes all teaching, research, and outreach programs. The program is administered by the Radiation Safety Officer (RSO) and his staff, as part of Radiological and Environmental Management (REM). The staff is equipped with radiation detection instrumentation to determine, control and document occupational radiation exposures at the reactor facility, and all laboratories using radioisotopes at the university under the By-product License 13-02812-04 (Broadscope).

Natural background radiation levels in the West Lafayette area result in an average exposure of about 100 mrem/yr. On the basis of normal reactor use, the maximum potential non-reactor room dose would be less than 1 mrem/yr, so there should be no significant contribution to the background radiation in unrestricted areas.

11.1.1 Radiation Sources

11.1.1.1. Reactor

Radiation from the reactor core if the primary source of radiation directly related to reactor operations. Radiation exposure rates from the reactor core are reduced to acceptable levels by the water in the pool and concrete shielding.

11.1.1.2. External Sources

Sources of radiation associated with reactor use include radioactive isotopes produced for research, activated components of experiments and activated samples.

11.1.2 Gaseous Effluents

The primary gaseous radionuclide considered to be produced in more than a negligible quantity by PUR-1 operations is Ar-41. This isotope is produced whenever air is in contact with a neutron radiation field. Naturally-occurring Ar-40, which comprises over 99% of all argon, undergoes a neutron capture reaction to produce Ar-41, which decays by beta and daughter product (K-41) gamma emission, with a half-life of 1.83 hours. Argon is found in air at slightly less than 1% concentration under STP conditions.

Smaller concentrations of gaseous radioisotopes will also be produced from other activation products in air, experimental procedures, and a slight possibility of very small quantities of fission product gases released into the reactor room environment from dissolved fission product gases in the pool water. However, the quantities of these other sources are very small compared to Ar-41 production.

The water in the reactor pool also contains dissolved air assumed that the dissolved air has an argon concentration equal found in atmospheric air. Some of this argon will activate and be released from the surface of the reactor pool into the reactor room air.

11.1.2.1. Argon Production in Experimental Facilities

Production of Ar-41 can occur in the dry drop tubes mounted near the core, and a movable drop tube that can be located near the core but is usually stored across the pool from the core. The production of Ar-41 can be estimated by:

The activity via the production rate of the ith radioiodine isotope in a plate is determined by the following:

$$A(t) = N\sigma\phi\left(1 - e^{-\lambda t_i}\right)e^{-\lambda t_d}$$

where

- N = total number of target atoms available for activation
- σ = microscopic absorption cross-section
- φ = neutron flux in neutrons/cm^2/s
- λ = decay constant
- t_i = irradiation time, and
- t_d = decay time.

For the most conservative evaluation, it is assumed that the quantity of Ar-41 reaches saturation, i.e., the irradiation time is 10 half-lives, or about 18.3 hours (Ar-41 $T_{1/2}$=1.83 hrs). PUR-1 is generally not operated more than four consecutive hours. It is also assumed that the Ar-41 is released immediately after irradiation, such that t_d=0. Thus the production equation simplifies to:

$$A = N \varphi \sigma$$

The Chart of the Nuclides[7] gives a value for σ(Ar-40) of 0.65 barns (6.5 x 10^{-25} cm^2) for the thermal neutron absorption capture cross-section. The number of Ar-40 atoms available for activation is a function of the volume of air in the experimental facility and the concentration of argon in air under STP conditions. Etherington[8] reports a concentration of 2.5 x 10^{17} atoms of argon per cubic centimeter of air under STP conditions. Using the isotopic abundance of 0.996 for Ar-40[7], a concentration of 2.49 x 10^{17} atoms of Ar-40 per cubic centimeter of air at STP is estimated.

There are four different drop tubes that can be used at PUR-1, two that are fixed and set against the reflector, one moveable drop tube, and one fixed tube located about 18 inches from the core. The moveable drop tube can be placed just outside the reflector, similar to the two that are

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[7] Baum, E. M., Knox, H. D., Miller, T. R., Knolls Atomic Power Laboratory. & Lockheed Martin. *Chart of the nuclides : nuclides and isotopes*. 16th edn, (KAPL : Lockheed Martin, 2002).

[8] Etherington, H. *Nuclear engineering handbook*. 1st edn, (McGraw-Hill, 1958).

fixed there. These tubes, and their associated fluxes (estimated from measured fluxes at 1000W extrapolated to 18 kW, or 12 kW+50% to account for all possible power uncertainties), are described in the table below.

Table 11-1

Tube	ID (cm)	Proximity to core	Estimated Neutron Flux at 18 kW (12 kW + 50%)
Small (SS)	1.55	Against reflector	2.79×10^8
Medium Aluminum	3.11	Against reflector	2.79×10^8
Large PVC	7.52	46 cm	8.79×10^5
Large moveable SS	12.08	Can be against reflector	2.79×10^8 max

Table 11-2

Tube	Effective Exposed Volume (cm^3)		Neutron Flux	Saturation Ar-41 Activity in Microcuries	Saturation Ar-41 Activity Production in Microcuries
Small (SS)	114.43	2.85E+19	2.79×10^8	0.14	1.47E-05
Medium Aluminum	463.68	1.15E+20	2.79×10^8	0.57	5.95E-05
Large PVC	2709.67	6.75E+20	8.79×10^5	0.01	1.10E-06
Large moveable SS	6982.02	1.74E+21	2.79×10^8	8.52	8.97E-04

11.1.2.2. Argon Production from Pool Water

Estimation of the Ar-41 production from dissolved air in the water of the reactor pool begins with a calculation of the exposure time of water passing through the core. Table 4-21 noted that the average coolant velocity in the through the core is 1.85 cm/second, assuming an 12 kilowatt operating power and natural convection through the core. The length of the active fuel channel is 60.96 cm, which gives a coolant transit time of 33.0 seconds, assuming the maximum velocity through the core. This is taken to be the exposure time of the water to the average flux throughout the core. Based on measurements of the peak thermal neutron flux in the core region at a 1 kilowatt power level, comparing to the MCNP5 calculation, and assuming linearity of thermal flux with reactor power, the peak thermal neutron flux in the core is assumed to be 1.66E11 n/(cm2*s).

The volume flow rate of water through the core is the product of the coolant velocity and the total flow area. Assuming a core with 13 standard fuel elements and 3 control rod fuel elements, the total flow area is the product of the flow area of an individual coolant channel and the total number of channels in the core. A standard fuel element has 15 flow channels, and a control element has 11. The total core flow area can be calculated to be 532.56 cm2, and the total volumetric flow rate is found to be 985.24 cm3/s.

The average out-of-core cycle time is given by:

$$T = V_p / \dot{V}$$

where

V_p = total pool volume, and

$\dot{V}$ = volumetric flow rate through the core

Assuming a pool volume of 6400 gallons (2.42×10^7 cm^3), and out-of-core cycle time of 2.46×10^4 seconds is obtained. This can be assumed to be the decay time for Ar-41 produced in the pool water.

The concentration of argon gas in the pool water can be predicted by Henry's Law. The dissolved concentration of a gas in contact with a liquid is proportional to the partial pressure of the gas and the temperature of the liquid. Dorsey[9] reports values for air at STP conditions in water that allow an estimation of 8.65×10^{15} atoms of Ar-40 per milliliter of water, assuming a water temperature of 25 degrees C. The saturation activity of Ar-41 in the pool water may then be predicted from:

$$A(t) = N\sigma\phi\left(1 - e^{-\lambda t}\right) \Big/ \left(e^{-\lambda[t+T]}\right)$$

where

N = concentration of Ar-40 atoms in the water
σ = microscopic absorption cross-section
φ = average neutron flux in neutrons/cm^2/s
λ = decay constant for Ar-41
t = exposure time of water in the core, and
T = average out-of-core cycle time.

Substituting all of the appropriate values into this equation, we find an estimate of 3.5 disintegrations/second/cm^3. Dividing this value by the decay constant yields an estimated density of 3.32×10^4 atoms of Ar-41/cm^3.

As water passes through the core it is heated slightly (~5 degrees), which reduces the solubility of air in the water. For this calculation, it is assumed that 25% (very conservative) of the dissolved argon is released from the water because of core heating. Some of this released argon will be re-dissolved as it mixes with cooler water in other regions of the pool.

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[9] Dorsey, N. E. *Properties of ordinary water-substance in all its phases: water-vapor, water, and all the ices*. (Reinhold Publishing Corporation, 1957).

Measurements done at other reactors allow an estimate of 50% re-dissolving fraction. Thus, the argon available for release to the building air is given by:

$$S_1 = F_1 F_2 N_{41} \dot{V}$$

where

F_1 = Ar-41 concentration in the water at equilibrium,
F_2 = release fraction from heating (assumed to be 25%),
N_{41} = redissolving fraction (assumed to be 50%), and
$\dot{V}$ = volumetric flow rate through the core.

Substituting the appropriate values in this equation leads to an available release term of 4.09 x 10^6 atoms of Ar-41/second. This represents one component of Ar-41 release from the water.

Another release term arises from the tendency of dissolved gas at the surface of a liquid to escape to the air across the water-air boundary. Estimating the magnitude of this release term requires calculation of an effective exchange coefficient for argon (exchange coefficient being the amount of gas in a unit volume exchanged at the surface per unit time per unit area).

Other reactor facilities have analyzed this problem and provide possible exchange coefficients that appear to cover a wide range. For example, analyzing the gas exchange at the liquid-gas boundary in terms of the diffusion coefficient of argon gas dissolved in water and the mean-square distance traversed by a molecule, an estimate of 2.35×10^{-3} cm/second is obtained. However, measurements made of the Ar activity in the pool water of a TRIGA Mark III and subsequent analysis of these data indicate an exchange coefficient of about 2.9×10^{-4} cm/second. Further, Dorsey[9] reports approximately equal surface exchange coefficients for gases such as air, O_2, and N_2. Assuming that the exchange properties of argon are similar to those of these gases, an exchange coefficient of about 5.7×10^{-3} cm/second is possible. Note that these estimates vary by almost a factor of 10.

In the interest of conservatism, the largest exchange coefficient (5.7 x 10^{-3} cm/second) is assumed in this calculation. Using this, the release rate from gas exchange at the surface of the pool is given by:

$$S_2 = 0.93B \cdot N_{41} A_S \tag{1}$$

N_{41} = Ar-41 concentration in the pool water
B = exchange coefficient, and
A_S = surface area of the pool (4.67×10^4 cm^2)

Using this equation a release rate of 8.23×10^6 atoms/second is obtained. Now, the total source term for Ar-41 released from the pool water is obtained by adding this to the previous estimate for dissolved argon:

$$\begin{aligned} S_{41} &= S_1 + S_2 \\ &= 4.09 \times 10^6 + 8.23 \times 10^6 \\ &= 1.23 \times 10^7 \text{ atoms/s} \end{aligned} \tag{2}$$

This is the source term for Ar-41 released from the pool water to be used later in estimating doses and isotopic concentrations. The source term assumes 12 kilowatt operation for a time sufficient to attain saturation activity.

11.1.3 Estimated releases in the Restricted Area

11.1.3.1. Types of Releases

Release of Ar-41 from experimental facilities can occur as either a puff from displacement of the air in a drop tube, or the surface of the pool, a continuous stream. Section 6.1.2 discussed the estimated source terms for Ar from experimental facilities and Ar from the surface of the pool, assuming 12 kilowatt operation. The following sections will analyze individual release scenarios and their consequences. These analyses concern releases made within the confines of the reactor room, which is defined as a restricted area.

11.1.3.2. Puff Release from the Drop Tubes

The Ar-41 activities in the drop tubes presented in Table XXXXX are very conservative, and it is not realistic for those to be achieved. Any theoretical release from those tubes might arise from displacement of the air in the tube by the insertion or removal of a sample with a diameter close to the inner diameter of the tube itself. Looking at the largest tube, a saturation activity of 8.52 microcuries is available for release. One could even consider the dispersal of the air containing Ar-41 from all of the tubes at once. The sum of the activities in all of the tubes in Table XXXXX is only 9.24 microcuries. If we assume instantaneous and ideal mixing of that activity to the reactor room air (4.24×10^8 cm^3), we achieve a concentration of 2.18×10^{-8} microcuries/cc of room air. Per 10 CFR 20, the derived air concentration limit for Ar-41 is 3×10^{-6} microcuries/cc. Thus, the Ar-41 concentration from dispersal of the air within the drop tubes to the reactor room is two orders of magnitude lower than the allowable limit. Furthermore, the exhaust fan will reduce that concentration quickly, changing the room air out in under two hours. Therefore we can conclude that Ar-41 release from the drop tubes will not endanger those working within the reactor room.

11.1.3.3. Continuous Release from the Pool Water

Section XXXXX estimated the release rate of Ar-41 from the pool water of 1.18×10^7 atoms/s. Multiplying this release rate by the decay constant for Ar-41 (1.05×10^{-4}/s), and converting this activity to microcuries yields an release estimate of 0.035 microcuries/second from the pool. Expressing the concentration buildup of the isotope

in the air of the Reactor Building, accounting for losses from radiological decay and building purging, leads to an equation similar in form to the production of a radioactive material by neutron irradiation, assuming a constant term for isotope production:

$$C(t) = P\left(1 - e^{-\lambda t}\right) \Big/ \lambda V$$

where

C(t) = time-dependent concentration of 41 Ar in the building air at time t after the reactor reaching full power

P = Release rate of Ar-41 from the pool

λe = decay constant based on the effective half life, Te, of Ar-41 in the reactor room (see below)

V = reactor room volume.

The effective half-life of the Ar-41, T_e, is based on the half-life from radioactive decay, and the half-life from room purging. The reactor room has a volume of 424 cubic meters, and the exhaust fan will move 0.220 cubic meters/s. This means that the room will purge in approximately 32.1 minutes. Assuming an equilibrium condition, we can assume that the half-life from room purging is 32.1 minutes. Therefore, the effective half-life is defined as follows:

$$T_e = (T_d \times T_p) / (T_d + T_p)$$

and the effective half-life of 24.84 minutes is obtained. By substituting the appropriate values into the equation for C(t), and assuming a time t that is sufficiently long to achieve equilibrium (not realistic), the resulting estimate for the concentration of Ar-41 in the room air released from the pool is 1.77×10^{-7} microcuries/cc. This is also well below the DAC for Ar-41 as listed in 10 CFR 20. Even with the addition of the Ar-41 released from the experiment drop tubes, the limit is not reachable such as to impair worker safety or violate 10 CFR 20.

11.1.4 Radiation Protection Program

Purdue University “Executive Memorandum No. B-14” establishes the University administrative structure for radiation protection, and a “Radiation Safety Manual” is published and maintained by REM, which contains the rules and radiation safety procedures for all laboratories using radioisotopes and/or ionizing radiation, including the reactor. Routine surveys are performed of the reactor room and include analysis of the reactor pool and reactor room air by personnel from REM.

The University has a variety of detecting and measuring instruments for monitoring potentially hazardous ionizing radiation. The instrument calibration procedures and techniques ensures that any credible type of radiation and any significant intensities will be detected promptly and measured correctly.

All reactor-related personnel are required to attend a radiation safety training session before they begin work at the reactor. Written procedures have been prepared that address routine health physics monitoring at the University’s research reactor facility, and all reactor personnel are trained in these as well.

11.1.5 ALARA Commitment

The University is committed to the principle of ALARA (As Low as Reasonably Achievable), and REM makes every effort to keep doses to a minimum. All unanticipated or unusual exposures are investigated.

13 ACCIDENT ANALYSES

In this chapter, details of the analysis of various accident scenarios are presented. The results of some of these analyses validate the safety system settings established in the Technical Specifications for the PUR-I. The potential effects of the accidents on the health and safety of the staff and public are analyzed.

13.1 Accident-Initiating Events and Scenarios

13.1.1 Maximum Hypothetical Accident

In the scenario of this accident it is assumed that a capsule containing irradiated fissile material breaks and a portion of the fission product inventory becomes airborne. The consequences of the release are analyzed for both the reactor staff and general public. Since the potential impact of this postulated accident is greater than in any other accident analyzed, the failure of a fueled experiment is designated as the maximum hypothetical accident of the PUR-I.

13.1.2 Insertion of Excess Reactivity

This accident scenario characterizes the reactor response to an insertion of the maximum allowable excess reactivity for the PUR-1 reactor, 0.6% Δk/k. This transient is examined in four different scenarios, a fast (step) and slow insertion, and each of these with and without scram. In the with scram case, it is assumed that the first scram signal (likely from short period) fails, and the reactor trips on power. The power scram is set for 120% power (12 kW), and it is also assumed that there is a 50% uncertainty in power measurement (actual scram happens at 18 kW), and that the scram signal is delayed by 0.1s before power is cut to the electromagnets that hold the shim-safety rods at their respective heights. It is also assumed that only the least reactive shim-safety rod, SS-2, is able to be inserted into the core (stuck rod scenario).

For the no scram case both rods are stuck at their fully raised position out of the core, which would be similar to the reactor being critical on one shim-safety rod, and that rod being ejected from the core (not a credible accident).

The results for these accidents are summarized in Table 13-1.

Table 13-1: Peak power and clad temperature for trip and no-trip insertions of 0.6% Δk/k.

SCRAM	Reactivity Inserted	P_0 (kW)	P_{max} (kW)	Time of Peak Power (s)	$T_{clad,max}$ (°C)		
					at t=0	at Peak Power	Maximum
YES	0.6% Δk/k step	10	40.3	0.187	49.94	49.73	49.94
YES	0.6% Δk/k over 10 s	10	18.4	7.83	49.94	42.30	49.98
NO	0.6% Δk/k step	10	2388	672	49.94	133.08	133.08

NO	0.6% Δk/k over 10 s	10	2388	680	49.94	133.08	133.08

13.1.3 Loss of Coolant

The reactor pool is designed to prevent the possibility of an unintentional drainage. It is constructed of steel and set in a second steel tank with the interstitial region filled with sand. The tank rests on a concrete pad about 15 feet below the floor of the Reactor Room, which is in the basement of the building. The pool has no drains. Therefore, a sudden loss of coolant is considered to be extremely remote. If the pool drained instantaneously, while the reactor was operating, the loss of water (moderator) would shut the reactor down.

The most severe problem identified in this accident scenario if the removal of decay heat during and after loss of coolant. There is no danger of significant fuel overheating as long as the core stays immersed and heat can be removed by the water. If the core were to become uncovered, heat transfer would occur by natural convection of ambient air. For this case, the amount of heat removed is proportional to the cladding temperature. According to commonly accepted models, the decay heat generation of a reactor is approximately 6.3% of the original operating power, and it drops off steadily as time goes on. Using this fraction, and applying it to the 12 kW operating power with an additional 50% power measurement uncertainty added in for a total of 18 kW, the power immediately after shutdown will be approximately 1098 W, with an average plate power of approximately 5.8 W. This heat rate will not be enough to lead to fuel damage.

In any reasonably conceivable accident scenario, the leakage of water from the reactor pool is expected to be rather slow. In such a case the radiation area monitor mounted directly above the core would detect any additional radiation coming from the core due to the decreasing pool water level. The pool water level is checked during daily routine operations. It is concluded, that a slow leak of pool water would be discovered early and specific actions could be taken to mitigate it consequences. It is concluded that no adverse consequences are to be expected to the health and safety of the public or the staff from this type of accident.

13.1.4 Loss of Coolant Flow

The Purdue University Reactor is cooled by natural convection, with peak flow rates at the onset of nucleate boiling (ONB, 98.11kW) of 5.41 cm/s, and nominal flow rates at 10 kW of 1.65 cm/s. The only consideration for a loss of coolant flow (LOCF) scenario would be a blockage in a channel. The power density of PUR-1 at 12 kW is such that the individual plate power is very low, and conduction of the heat to adjacent channels would occur, and plate temperatures would remain well below ONB temperatures of 112°C, which is still more than 400°C below the safety limit for the fuel. Therefore this accident is not analyzed.

13.1.5 Mishandling or Malfunction of Fuel

Operation of PUR-1 produces an inventory of radioisotopes in each fuel plate resulting from the fission of 235U. Section 13 of NUREG 1537 suggests that an analysis should be performed for a maximum hypothetical accident (MHA) involving a release of fission products that would have consequences greater than any credible accident. For a low-power (< 2 MW) MTR fuel reactor, the recommended analysis is that resulting from cladding stripped from one face of one fuel

plate. In this section, the consequences of an event in which fission fragment radioisotopes are released from a fuel plate to the pool water and subsequently to the atmosphere of the building, ultimately to be released to the outside air, will be analyzed.

Such an event is extremely unlikely during routine operation of the PUR-1. The core operates in a natural convection cooling mode, so failure of a fuel plate as a result of hydraulic pressures in the core is not likely. Previous analyses show that the average coolant velocity in an average channel is 1.8 cm/sec, with 1.93 cm/sec being the coolant velocity in the hot channel. Such coolant velocities are low enough that significant pressures will not be generated anywhere in the core, nor will excessive wear or erosion of the fuel plate surface occur.

Fuel plate cladding failure can occur as a result of corrosion action over years. However, previous studies of MTR-type fuel plates have shown that a cladding hole on the order of several square centimeters in size must be present before significant amounts of radioactivity can be detected in the pool water. It is very unlikely that a hole of such size will form suddenly during normal operations. Current operating procedures call for periodic testing of the pool water to determine radioisotopic content on a regular basis. In addition, fuel element inspections are performed annually. Each year some of the fuel elements are inspected such that all of the elements are inspected at least once in a five year period. During the operating history of the HEU-fueled PUR-1 (over 45 years), no evidence of significant fuel plate surface corrosion (to the point of showing major defects such as formation of thick oxidation layers, fracture defects, or stress lines) has been found.

Fuel element maneuvers are always conducted in the reactor pool. They are removed from the core and moved into the storage racks, one at a time, using a hand-held fuel handling tool. Annually a fuel element is removed from the pool for inspection. A fuel element weighs about 4.5 kg (9.9 lb) in air and only about 2.8 kg (6.2 lb) in water. Therefore, even if a fuel element should fall from the handling tool during its transfer it is not heavy enough to cause any considerable damage. The most severe consequence likely to occur would be some denting of the end fittings since the fuel element, being an elongated object, would tend to fall in water in a rather upright position.

The PUR-I Standard Operating Procedures define administrative steps which are intended to prevent a fuel handling mishap. They are:

1. All fuel handling is done in accordance with written procedures.

2. Loading operations are done by qualified personnel under direct supervision of a Senior Operator.

3. The fuel handling tool is kept locked with the key secured to prevent unauthorized movement of fuel. It is concluded that no adverse consequences are to be expected to the health and safety of the public or the staff from this type of accident.

Even in light of all of this, to provide a bound on the consequences of any hypothetical release of fission products, an analysis was performed in subsequent sections.

13.1.6 Experiment Malfunction

In this section an analysis is performed to assess the hazard associated with the failure of an experiment in which fissile material has been irradiated in the reactor. In the scenario of this

accident it is assumed that a capsule containing irradiated fissile material breaks and a portion of the fission product inventory becomes airborne. The consequences of the release are analyzed for both the reactor staff and general public. Since the potential impact of this postulated accident is greater than in any other accident analyzed, the failure of a fueled experiment is designated as the maximum hypothetical accident of the PUR-I. This accident is analyzed in Section 13.2.1.

The flooding of the 12.7 cm drop tube when placed in position G6 as described in the SER NUREG-1283 resulted in a reactivity insertion of 0.246% Δk/k, which is below the insertion resulting from the failure of a moveable experiment examined in the conversion proposal. This accident scenario is within the envelope of the examined accident cases, and therefore was not analyzed.

13.1.7 Loss of Normal Electrical Power

The loss of normal electric power at PUR-1 will shutdown the reactor by simulating a scram with a loss of power to the electromagnets that hoist the shim-safety rods. This action will shutdown the reactor from any conceivable operating condition. Since the reactor is cooled by natural convection, there are no shutdown and decay heat issues. There is adequate heat capacity in the reactor pool to address shutdown and decay heat loads. Therefore, this accident scenario is not addressed.

13.1.8 External Events

13.1.8.1. Fire or Explosion

The reactor building is a steel frame structure with concrete block and brick construction. Additionally, the reactor pool is located below ground level, and the reactor itself below the floor level in the reactor room. The materials surrounding the reactor core are concrete, steel and earth. There is limited combustible material in the reactor room, but it is virtually impossible to exclude all burnable materials. Therefore, some small fire possibility exists. There is portable fire-fighting equipment available, and PUR-1 staff members are trained in use of that equipment. Purdue Fire Department is located on campus, and is available for assistance 24 hours a day.

In general, personnel judgment is used in deciding the response to a fire. If the fire is small and can be easily controlled, little action is necessary to safely shutdown the reactor and control the fire. If the fire is of larger magnitude, the reactor will be secured and outside assistance obtained.

Because of the large volume of water surrounding the core, damage to the fuel is not likely even in a severe fire scenario. Even if the fire involved the control systems, power to those circuits could be cut at the wall of the reactor room, and the control rods would drop and safely shut down the reactor.

Explosions external to the building could affect the reactor and its systems; however explosions external to the reactor pool will have little effect on the reactor due to its location and the construction of the pool. The amount of water covering the core will aid in the reduction of risk to pyrotechnic devices thrown into the pool, either serving to quench the devices, or absorb most of the energy from the detonation.

With these facts, and the location of the reactor in the pool, damage from fire or explosion is not likely.

13.1.8.2. Acts of Sabotage

The reactor room is always kept locked except when personnel or materials access is needed. There is a security plan in place, and security systems designed to protect all of the controlled areas in the reactor facility. Purdue Police, a fully functioning police department with arrest authority, is available for assistance, and is staffed 24 hours a day. The PUR-1 Security Plan is reviewed and approved by the U.S. NRC, and provides for in-house procedures to ensure facility protection.

Riots or acts of civil disobedience directed against the reactor facility would be recognizable is sufficient time to ensure safe shutdown of the reactor. Personnel at the reactor could quickly alert local law enforcement, and quickly come under their protection. Then access to the area could be restricted until it is safe to resume normal operations.

13.2 Accident Analysis and Determination of Consequences

13.2.1 Maximum Hypothetical Accident (Mishandling or Malfunction of Fuel)

Section 13 of NUREG 1537 suggests the maximum hypothetical accident for a low power (<2 MW) MTR reactor is the cladding being stripped from the face of one fuel plate. This accident is not seen as credible, but would serve as a bounding case for partial breakage of a fuel plate, or failure of a plate in water with mixing of the fission products in the pool with eventual release to the reactor room air.

For this analysis, the highest power place is chosen from the analysis in Table 4-8. Plate 1348 has a projected plate power of ~97 W at 12 kW, and if a 50% power uncertainty is added to envelope all possible power uncertainties, the plate power for 1348 becomes 145.5 W.

If the plate cladding is stripped, it is the gaseous fission products that will be the main concern, and those will be the only radionuclides analyzed here. Further simplifying assumptions are as follows:

1. All of the fission products are assumed to be at saturation, which assumes an infinite operation at the specified power.

2. the fuel plate which fails does so at the end of this operation,

3. the failed fuel plate is located at the peak flux point in the core,

4. the released radionuclides are perfectly mixed with the pool water at the time of release, to be transferred by diffusion to the building air,

5. the primary radionuclide of consequence is elemental iodine.

13.2.1.1. Source Term Estimation for Radioiodine Release

This section will estimate the total amount of radioiodine released from the failed fuel plate. No credit is taken for the reduction in activity resulting from radioactive decay during the time of the

release, i.e. an instantaneous release of the radioiodine that can escape the fuel is assumed. Complete and perfect mixing of the available radioiodine inventory with the reactor pool water is also assumed.

The activity via the production rate of the *i*th radioiodine isotope in a plate is determined by the following:

$$A_i = \lambda_i N_i = KPF_i$$

where

- K = conversion constant = 3.1×10^{10} fissions/second/watt
- P = plate power in watts
- F_i = fractional fission yield for the i^{th} radioiodine
- λ_i = decay constant for the i^{th} radioiodine, and
- N_i = saturation number of atoms of the i^{th} radioiodine produced

The constants F_i, λ_i, and the results of calculations for $\lambda_i N_i$ and Ni for plate 1348, are shown in Table 13-2 assuming 18 kW core operating power (to incorporate the aforementioned uncertainties). The total number of radioiodine atoms available in the the plate is 1.764×10^{17}. Assuming that the iodine forms the I_2 molecule, the total number of I_2 molecules is thus 8.82×10^{16}.The calculated values for each of the five iodine isotopes of concern are shown in Table 13-2.

Table 13-2: Values and Results for Radioiodine Production in PUR-1 Plate 1348

Isotope	F_i	$T_{1/2}$ (s)	λ_i (s^{-1})	$A_i=\lambda_i N_i$ (dis/s)	N_i (atoms)
I-131	0.029	6.934E+05	9.997E-07	1.308E+11	1.308E+17
I-132	0.043	8.262E+03	8.390E-05	1.940E+11	2.312E+15
I-133	0.065	7.499E+04	9.243E-06	2.932E+11	3.172E+16
I-134	0.08	3.150E+03	2.200E-04	3.608E+11	1.640E+15
I-135	0.064	2.369E+04	2.926E-05	2.887E+11	9.865E+15
				TOTAL	**1.764E+17**
				I_2	**8.82E+16**

It is assumed that not all of the iodine produced will be released from the plate. As suggested NUREG/CR-2079, only the fission fragment gases within recoil range of the surface of the fuel (1.37×10^{-3} cm for aluminum matrix fuels) will escape in this scenario. The thickness of the fuel meat in a PUR-1 plate is 0.0508 cm. Therefore the fraction of the fission product gas release is given by:

$$f = \frac{1.37 \times 10^{-3}\ \text{cm}}{0.0508\ \text{cm}} = 0.0207$$

Therefore, Table 13-2 can be revised as shown in Table 13-3.

Table 13-3: Radioiodines released from Plate 1348 into pool water.

Isotope	F_i	$T_{1/2}$ (s)	λ_i (s-1)	$A_i=\lambda_i N_i$ (dis/s)	N_i (atoms)	N_i (atoms) Released
I-131	0.029	6.934E+05	9.997E-07	1.308E+11	1.308E+17	2.709E+15
I-132	0.043	8.262E+03	8.390E-05	1.940E+11	2.312E+15	4.785E+13
I-133	0.065	7.499E+04	9.243E-06	2.932E+11	3.172E+16	6.566E+14
I-134	0.08	3.150E+03	2.200E-04	3.608E+11	1.640E+15	3.394E+13
I-135	0.064	2.369E+04	2.926E-05	2.887E+11	9.865E+15	2.042E+14
				TOTAL	1.764E+17	3.651E+15
				I_2	8.82E+16	1.826E+15

Therefore, a total of 1.826×10^{15} radioiodine molecules are expected to be released to the reactor pool water. The reactor pool contains about 6400 gallons of water. With the assumption of perfect and complete mixing of the radioiodine in the water, the mole fraction in the pool water is calculated from:

$X_W = (N_I/N_A) / (VK\rho/M)$ = mole fraction of radioiodines in the water

where

N_I = number of radioiodine molecules released
N_A = Avogadro's Number (6.023×10^{23}/mole)
V = Reactor pool volume (6400 gallons)
K = conversion factor (3.8×10^3 cm^3/gallon)
ρ = density of water (1 gram/cm^3)
M = molecular weight of water (18 grams/mole)

Substituting these values in the equation gives a mole fraction of 2.24×10^{-15} for the radioiodine. The partial pressure of radioiodine in air, P_I, can be estimated by:

$P_I \quad P_o X_W$

with the value of X_W from above, and Po being the vapor pressure of pure iodine, which can be estimated by:

$\log_{10}(P_o) = AC/T + B$

where

P_o = vapor pressure of pure iodine in mm of Hg

A = molar heat of vaporization (13057 cal/mol for I_2)

C = constant = -0.2185

T = bulk pool temperature in degrees Kelvin

B = constant = 9.24 (for I_2)

Assuming a bulk pool temperature of 30°C (303 K), a value of 0.32 mm Hg is obtained for P_o. Further substitution yields an estimate for P_I of 1.50×10^{-15} mm Hg for the partial pressure of the radioiodine in air. From this, the molar fraction of radioiodine in air, assuming equilibrium at standard atmospheric pressure, is given as:

$X_{air} = 1.50 \times 10^{-15}$ mm Hg / 760 mm Hg

$= 1.97 \times 10^{-18}$.

Assuming the free volume, V_r, of the reactor room is approximately 424 m3 (4.24×10^5 L), the total moles of radioiodine present can be estimated as:

$M_I = X_{air} V_r / (24.5 \text{ liters/mole})$

which yields an estimate of 3.41×10^{-14} moles.

It is now possible to estimate the number of moles of the i^{th} radioiodine, denoted as M_i, from the estimates shown in Table 13-3 for the relative populations of the relevant iodine isotopes. These values are given by:

$$M_i = M_I \cdot \frac{N_i}{N_p}$$ where N_p is the total iodine atoms released from plate 1348.

The activities of the i^{th} radioidine isotope in the building air is then calculated from:

$A_i = 2 \cdot M_i \cdot N_A \cdot \lambda_i$

The results are shown in Table 13-4. Now it is possible to estimate the quantity of a given radioiodine present at time *t* after the rupture of plate 1348, by using:

$S_i = S_i(0) e^{-\lambda_i t}$

where

λ_i decay constant of the ith iodine

t = time following iodine release

Si(0) = initial quantity present

For this analysis, we will assume that the quantity S_i is expressed in units of dose (rads) to the thyroid gland of a person breathing the radioiodine-bearing air. Although the concentration of iodine in the building air is continuously reduced by various processes such as radioactive decay, purging of the building air by the exhaust fan, and plating out of the iodine on surfaces, we will assume only a reduction in concentration resulting from radioactive decay. Essentially, this assumes that the releases from the pool surface are balanced by losses other than radioactive decay, and equilibrium is established between the pool water radioiodine and that in the building air. This also implies that the concentrations of iodine in the air outside the building,

taking no credit for dissipation in the air outside the building, will be the same as those in the building. It is believed that these assumptions are conservative.

The concentration of a given radioiodine in units of thyroid dose per unit volume of air is given as:

Ci = Si/Vr

where Vr is the free air volume of the reactor room. The results are shown in Table 13-4.

Table 13-4: Number of Moles and Activities of Radioiodine in Reactor Room Air After Release from the Failed Plate

Isotope	M_i (mol)	λ_i (s^{-1})	$A_{i(0)}=\lambda_i N_i$ (dis/s)	$A_{i(0)}$ (Ci/cm^3)
I-131	2.53E-14	9.997E-07	8.23E-07	1.94E-15
I-132	4.47E-16	8.390E-05	1.22E-06	2.88E-15
I-133	6.13E-15	9.243E-06	1.84E-06	4.35E-15
I-134	3.17E-16	2.200E-04	2.27E-06	5.35E-15
I-135	1.91E-15	2.926E-05	1.82E-06	4.28E-15

13.2.1.2. Thyroid Dose Consequences

The integrated thyroid dose to a person breathing the reactor room air containing radioiodine contaminate for a time period from 0 to time T is given by:

$$D_i = \int_0^T B \cdot C_i dt = \left[\frac{B \cdot S_i(0)}{V_b}\right] \int_0^T e^{-\lambda t} dt = \frac{B \cdot S_i(0)}{V_b} \cdot \left(1 - e^{-\lambda t}\right)$$

Integrating the expression over the given limits yields

$$D_i = \frac{B \cdot S_i(0)}{V_b \lambda_i}\left(1 - e^{-\lambda T}\right)$$

where B is the standard breathing rate, assumed to be a constant of 3.47×10^{-4} m^3/s (347 cm^3/s). The integrated thyroid dose estimates for varying exposure times are then given as shown in Table 13-5.

If it is assumed that the fission products are instantaneously released and uniformly distributed in the Reactor Room air, and the free volume of the Reactor Room is approximately 4.24×10^8 cm^3 (4.24×10^5 liters). The quantity of a given quantity of radioiodine present at time "t" after the fuel plate rupture is estimated by

$$A_{i,t} = A_{i,(0)} e^{-\lambda_i t}$$

The dose to the thyroid due to inhaled radioiodines can be estimated by multiplying the activities of the isotopes by the Dose Conversion Factors (DCFs) for the respective nuclides (as given in Reg Guide 1.25) or

$$S_i = DCF_i \bullet A_i .$$

And if follows that the concentration of a given radioiodine in units of thyroid dose per unit volume of air is thus

$$C_i = S_i / V_b$$

Estimates of the respective activities and concentrations of the radioiodines are shown in Table 13-3, and estimates of the thyroid dose rates are shown in Table 13-5.

Table 13-5: Thyroid Dose Rates

Isotope	$A_i(0)$ (Ci/cm3)	DCF_i (rads/Ci)	$C_i(0)$ (rads/cm3)
I-131	1.94E-15	1.48E+06	2.87E-09
I-132	2.88E-15	5.35E+04	1.54E-10
I-133	4.35E-15	4.00E+05	1.74E-09
I-134	5.35E-15	2.50E+04	1.34E-10
I-135	4.28E-15	1.24E+05	5.31E-10

Table 13-6: Integrated thyroid dose estimates for several exposure periods following release of Plate 1348 radioiodines into the reactor pool.

			Dose in mrem for Several Exposure Periods					
Isotope	**$S_{i(0)}$ (rads/cm^3)**	**λ_i (s^{-1})**	**90 sec**	**2 hours**	**1 day**	**2 days**	**7 days**	**30 days**
I-131	2.87E-09	9.997E-07	0.09	7.15	82.52	158.21	452.45	922.51
I-132	1.54E-10	8.390E-05	0.00	0.29	0.64	0.64	0.64	0.64
I-133	1.74E-09	9.243E-06	0.05	4.21	35.94	52.11	65.09	65.33
I-134	1.34E-10	2.200E-04	0.00	0.17	0.21	0.21	0.21	0.21
I-135	5.31E-10	2.926E-05	0.02	1.20	5.80	6.26	6.30	6.30
			0.17	**13.01**	**125.10**	**217.42**	**524.69**	**994.99**

As can be seen from these results, about one month of continuous exposure to released radioiodine in the reactor room air would be required to attain a thyroid dose equivalent of ~1 rem. Even this exposure would be extremely unlikely, since it is difficult to conceive of a credible combination of accident conditions and personnel occupancy which will result in such doses being achieved. The imposition of such limited cloud dispersion effects required to approach these estimates is not realistic. It is more likely that dispersive effects will result in much lower doses. For example, even the building blower exhaust at 424 CFM volume flow rate will cause a

concentration reduction. This dispersion will also be enhanced by natural dispersive effects such as wind speed.

13.2.1.3. MHA considering All Gaseous Fission Products

A similar analysis can be done for all gaseous radionuclides released from a single ruptured fuel plate. The same analysis for iodine as was done in the preceding sections applies, as well as all noble gases (fission products) available in the plate released to the pool water and then to the building air. As before, we will take no credit for decay of the radioisotopes during release and dispersion. This assumption, as before, leads to conservative results in that the estimates obtained are higher than those that would actually occur in this postulated accident.

13.2.1.4. Whole-Body Gamma Dose Estimation

As with the radioiodine release model considered in previous sections, we will assume that the released radionuclides are dispersed into a hemispherical cloud, perfectly mixed with the air in this hemispherical volume. The dose consequences to a person submersed in this cloud are considered.

For a submersion dose, Cember [7] recommends calculating the dose from an infinite hemisphere of gas. To accomplish this, first we calculate the dose from an infinite cloud and then divide by two to account for an infinite hemispherical geometry. By assuming an infinite cloud, we can assume that the density of absorbed energy is equal to the density of emitted energy. This allows us to do a straight-forward calculation from activity density to dose rate. For an isotope i:

$$\dot{D}_i = \frac{A_i}{\rho V} e^{-(\lambda_i + \Lambda)t} E_i F_1 F_2 F_3 F_4 F_5 F_6$$

where

- Ai = source term activity in Ci
- ρ = density of air = 1.293 kg/m3
- V = reactor room volume
- λi = nuclide decay constant in s-1
- Λ = building leakage volume in s-1
- t = time after release in seconds
- Ei = gamma energy in MeV per disintegration
- F1 = factor to convert Ci to dis/s
- F2 = factor to convert MeV to J = 1.6x10-13
- F3 = factor to convert J/kg to rad = 100
- F4 = factor to convert rad to rem (Q-factor) = 1
- F5 = factor to account for stopping power of tissue ≈1.1
- F6 = factor to account for hemispherical geometry

The term $e^{-(\lambda_i+\Lambda)t}$ accounts for reduction in the source term over time due to both radioactive decay and leakage from the building. If we integrate this equation from the release time t=0 to the final time t=T, we get a dose for that period of exposure.

$$\dot{D}_i = \frac{A_i}{\rho V} \frac{e^{-(\lambda_i+\Lambda)t}}{-(\lambda_i+\Lambda)} E_i F_1 F_2 F_3 F_4 F_5 F_6$$

Information relevant to these calculations is shown in Table 13-8. Also, in Table 13-8, the value for the activity $A_i = \lambda_i N_i$ is calculated as follows:

$A_i = \lambda_i N_i = KPF_i$

where

- K = conversion constant = 3.1×10^{10} fissions/second/watt
- P = plate power in watts
- F_i = fractional fission yield for the i^{th} radioisotope
- λ_i = decay constant for the i^{th} radioisotope, and
- N_i = saturation number of atoms of the i^{th} radioisotope produced

Values for the isotopes in question are shown in **Error! Reference source not found.**. Dose results for the isotopes in question and for a number of different lengths of time can be seen in Table 13-9 and Table 13-10. Table 13-9 has results for the purge fan turned off, and Table 13-10 has results for the purge fan turned on.

Table 13-7: Calculation results for gaseous fission products released from the failed fuel plate.

Isotope	Fission Yield	λ_i (s^{-1})	Activity Released to pool	Isotope Activity in Reactor Room Air (C_i)	Gamma Energy (MeV/dis)
I-131	0.029	9.997E-07	7.318E-02	8.23E-07	0.4
I-132	0.043	8.390E-05	1.085E-01	1.22E-06	2.12
I-133	0.065	9.243E-06	1.640E-01	1.84E-06	0.55
I-134	0.08	2.200E-04	2.019E-01	2.27E-06	1.25
I-135	0.064	2.926E-05	1.615E-01	1.82E-06	1.50
Kr-85^m	0.013	4.298E-05	3.280E-02	3.28E-02	0.19
Kr-87	0.025	1.514E-04	6.309E-02	6.31E-02	0.63
Kr-88	0.036	4.068E-03	9.084E-02	9.08E-02	2.18
Xe-131^m	0.029	6.776E-07	7.318E-02	7.32E-02	0.002
Xe-133^m	0.065	3.650E-06	1.640E-01	1.64E-01	0.006
Xe-133	0.065	1.529E-06	1.640E-01	1.64E-01	0.08
Xe-135^m	0.064	7.556E-04	1.615E-01	1.62E-01	0.15

Xe-135m	0.064	2.107E-05	1.615E-01	1.62E-01	0.24

Table 13-8: Associated photon information for the gaseous fission products.

isotope	Avg. Gamma Energy (MeV)	Linear Attenuation Coefficient in Air (m-1)
I-131	0.4	3.90×10^{-3}
I-132	0.8	3.70×10^{-3}
I-133	0.55	3.90×10^{-3}
I-134	1.3	3.40×10^{-3}
I-135	1.5	3.30×10^{-3}
Kr-85m	0.2	3.50×10^{-3}
Kr-87	2	3.00×10^{-3}
Kr-88	2	3.00×10^{-3}
Xe-131m	0.16	3.30×10^{-3}
Xe-133m	0.23	3.60×10^{-3}
Xe-133	0.08	3.20×10^{-3}
Xe-135m	0.52	3.90×10^{-3}
Xe-135m	0.25	3.60×10^{-3}

These values that were obtained assuming an infinite cloud set a very conservative upper bound on the dose. If the air volume of the reactor building is considered to be in the shape of a hemisphere, this hemisphere would have a radius of about ten meters. The gammas coming from the nuclides of interest have an average path length in air on the order of 250 meters. This indicates that the infinite cloud assumption is grossly overestimating the actual dose, so we need to use a factor to correct for the non-infinite extent of the cloud.

To obtain this factor, we can think of a spherical cloud consisting of a number of thin concentric shells. Consider a shell at an arbitrary distance 'r' from the origin, which is our point of interest. The contribution of gammas from a point on this shell passing through the origin is:

$$\varphi = \frac{S}{4\pi r^2} e^{-\mu r}$$

where

- φ = flux and the origin
- S = volume-distributed source (Ci/m3)
- μ = gamma absorption coefficient in air (m-1)
- r = distance from origin (m)

This equation accounts for the spread of the radiation away from the point as well as the attenuation from interactions with air. If we sum over all the points on this shell we get a contribution of:

$$\varphi = Se^{-\mu r}$$

Now, by integrating these thin shells over a radius of zero to infinity, we find the quantity of gammas passing through the origin from the infinite cloud.

$$\varphi = \int_0^{\infty} Se^{-\mu r} dr = \frac{S}{\mu}$$

Likewise, integrating from zero to the radius of our hemisphere 'R', we can find the relative quantity of gammas from the finite cloud.

$$\varphi = \int_0^{R} Se^{-\mu r} dr = \frac{S}{\mu}\left[1 - e^{-\mu r}\right]$$

Taking the ratio of these two quantities gives our correction factor.

$$F = 1 - e^{-\mu R}$$

If we take a Taylor expansion of this result and discard the higher-order terms (since μR is small), we get a correction factor of μR. Refer to Table 13-8 for values of the absorption coefficient for the different isotopes. Using this correction factor on the data from Table 13-9 and Table 13-10 gives the dose information seen in Table 13-11 and Table 13-12. These values are not as overly conservative as those found from the infinite-cloud assumption, but they are still conservative due to the approach used to estimate the source term underlying this analysis.

The results show that doses can be kept low to persons inside the building if exposure times are reduced. Thus, room evacuation is an appropriate response to this postulated event. A 90 second evacuation time is reasonable. Both evacuation and shutdown of ventilation systems are part of the emergency response procedures for PUR-1 operation, and form a part of the overall PUR-1 emergency plan. However, even for prolonged exposures to this release, integral whole-body doses can be expected to be lower in actual experience because of the conservative assumptions made in this analysis.

Table 13-9: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Infinite Cloud and a Leakage Fraction of 0.005 Hr^{-1} (Exhaust Fan Off)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-132	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-134	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-135	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-85m	1.01	1.85	2.54	3.98	5.26	5.80	5.86	5.86
Kr-87	6.35	11.44	15.51	23.47	29.67	31.73	31.89	31.89
Kr-88	19.01	23.71	24.88	25.25	25.26	25.26	25.26	25.26
Xe-131m	0.02	0.04	0.06	0.10	0.13	0.15	0.15	0.15
Xe-133m	0.16	0.30	0.41	0.65	0.87	0.97	0.99	0.99
Xe-133	2.14	3.94	5.44	8.64	11.62	13.01	13.20	13.20
Xe-135m	3.55	5.92	7.51	9.75	10.62	10.70	10.70	10.70
Xe-135m	6.31	11.57	15.94	25.15	33.54	37.27	37.74	37.74
	38.56	58.77	72.29	97.00	116.97	124.90	125.79	125.79

Table 13-10: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Infinite Cloud and a Leakage Fraction of 1.87 Hr^{-1} (Exhaust Fan On)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-132	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-134	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-135	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-85m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-87	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01
Kr-88	0.06	0.06	0.06	0.06	0.06	0.06	0.06	0.06
Xe-131m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-133m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-135m	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01
Xe-135m	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01
	0.10	0.10	0.10	0.10	0.10	0.10	0.10	0.10

Table 13-11: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Finite Cloud and a Leakage Fraction of 0.005 Hr^{-1} (Exhaust Fan Off)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-132	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001
I-133	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-134	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001
I-135	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001
Kr-85m	0.0354	0.0647	0.0889	0.1393	0.1840	0.2029	0.2050	0.2050
Kr-87	0.1905	0.3431	0.4652	0.7042	0.8900	0.9519	0.9566	0.9566
Kr-88	0.5703	0.7114	0.7463	0.7576	0.7577	0.7577	0.7577	0.7577
Xe-131m	0.0008	0.0014	0.0020	0.0032	0.0043	0.0048	0.0049	0.0049
Xe-133m	0.0058	0.0106	0.0147	0.0233	0.0313	0.0350	0.0355	0.0355
Xe-133	0.0686	0.1260	0.1741	0.2764	0.3720	0.4164	0.4224	0.4224
Xe-135m	0.1385	0.2311	0.2929	0.3802	0.4141	0.4173	0.4174	0.4174
Xe-135m	0.2272	0.4164	0.5740	0.9055	1.2075	1.3418	1.3587	1.3587
	1.2371	1.9048	2.3582	3.1899	3.8610	4.1282	4.1584	4.1584

Table 13-12: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Finite Cloud and a Leakage Fraction of 1.87 Hr^{-1} (Exhaust Fan On)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-132	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-133	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-134	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-135	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Kr-85m	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
Kr-87	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004
Kr-88	0.0019	0.0019	0.0019	0.0019	0.0019	0.0019	0.0019	0.0019
Xe-131m	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Xe-133m	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Xe-133	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
Xe-135m	0.0003	0.0003	0.0003	0.0003	0.0003	0.0003	0.0003	0.0003
Xe-135m	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004
	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032

Doses to persons outside the building will come from submersion in a cloud of released radionuclides and from radiation emitted from the reactor building. The submersion dose results from the diluted radionuclide stream from the exhaust fan or from natural flow of air through the building that exits at the roofline (if the exhaust fan has been shut off). An analysis for the activity concentration released from the building can be performed using the equation below, which includes release from the exhaust fan.

$$A_D = A \cdot q \cdot \Psi(x)$$

where

AD = effective exposure concentration in curies/m3
q = building exhaust rate in m3/second
Ψ(x) = dilution factor at distance x, in sec/m3
A = activity concentration in the exhaust stream

The dilution factor, Ψ(x), is calculated for the leeward side of the building (x=0), and assumes the release is made from the roofline of the building. It is further assumed that the wind velocity is steady at the time of the release, and is equal to 1 m/s. Thus, the dilution factor can be written as:

$$\Psi(x) = 1 / [(0.5) (s) (u)]$$

where

u = wind velocity in m/s
s = building cross sectional area normal to the wind direction (in m^2)

Assuming the prevailing winds are blowing at the time of the release, the cross sectional area of the building is 288 m^2. Making the appropriate substitutions, Ψ(0) is found to be 6.94×10^{-3} s/m^3 for a release from the roofline of the building. Using this value for Ψ(0) with the appropriate values for building exhaust rate gives values for activity concentrations outside the restricted area. These concentrations can then be used to calculate estimates for accumulated doses from the nuclides of interest as was done for immersion dose inside the building. For isotope 'i',

$$D_i = \frac{A_i \cdot q \cdot \Psi}{\rho V} \cdot \frac{1-e^{-(\lambda_i+\Lambda)t}}{} E_i \cdot F_1 \cdot F_2 \cdot F_3 \cdot F_4 \cdot F_5 \cdot F_6$$

where all variables are as defined previously in this section.

Table 13-13 shows the results of this calculation for submersion outside the building with the exhaust fan on. This can be considered to be a bounding case, and the dose rate is still below 10 CFR 20 limits for dose to the public. Turning the fan off would reduce the dose by more then three orders of magnitude. Even if the exhaust fan were left running, accumulated dose to persons outside would be minimal, particularly after the first day.

To calculate the direct dose from the reactor building to someone standing at ground level outside the building and restricted area, assume a half-hemisphere with a volume equivalent to that of the reactor room. The direct dose can then be calculated as a submersion dose from a finite hemisphere divided by two, since the dose only comes from half of a hemisphere.

$$D_i = \frac{A_i}{\rho V} \cdot \frac{1-e^{-(\lambda_i+\Lambda)t}}{\lambda_i+\Lambda} E_i \cdot F_1 \cdot F_2 \cdot F_3 \cdot F_4 \cdot F_5 \cdot F_7 \cdot \mu \cdot R$$

.

where

F7 = factor to account for half-hemispherical geometry = ¼

μ = gamma absorption coefficient in air (m^{-1})

R = radius of half hemisphere (m)

and all other variables are as defined above.

The results of this calculation with the exhaust fan off are shown in Table 13-14. This estimate is very conservative in that it assumes that a person is standing up against a building wall for an extended period of time, and it does not take into account absorption from building walls or concrete in the building. Therefore the hypothetical dose at this boundary will be estimated to determine if it is within 10 CFR 20 limits.

Table 13-13: Integral Whole-Body Gamma Doses From Submersion Outside of the Restricted Area Assuming a Leakage Fraction of 1.87 Hr^{-1} (Exhaust Fan On)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-132	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-133	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-134	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-135	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Kr-85m	0.0015	0.0028	0.0039	0.0061	0.0080	0.0089	0.0089	0.0089
Kr-87	0.0097	0.0175	0.0237	0.0358	0.0453	0.0484	0.0487	0.0487
Kr-88	0.0290	0.0362	0.0380	0.0386	0.0386	0.0386	0.0386	0.0386
Xe-131m	0.0000	0.0001	0.0001	0.0001	0.0002	0.0002	0.0002	0.0002
Xe-133m	0.0002	0.0005	0.0006	0.0010	0.0013	0.0015	0.0015	0.0015
Xe-133	0.0033	0.0060	0.0083	0.0132	0.0177	0.0199	0.0202	0.0202
Xe-135m	0.0054	0.0090	0.0115	0.0149	0.0162	0.0163	0.0163	0.0163
Xe-135m	0.0096	0.0177	0.0243	0.0384	0.0512	0.0569	0.0576	0.0576
	0.0589	0.0897	0.1104	0.1481	0.1786	0.1907	0.1920	0.1920

Table 13-14: Integral Whole-Body Gamma Doses From Direct (From the Building) Dose Assuming a Leakage Fraction of 0.005 Hr-1 (Purge Fan Off)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-132	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-134	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-135	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-85m	0.01	0.02	0.03	0.04	0.05	0.06	0.06	0.06
Kr-87	0.05	0.10	0.13	0.20	0.25	0.27	0.27	0.27
Kr-88	0.16	0.20	0.21	0.22	0.22	0.22	0.22	0.22
Xe-131m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-133m	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.01
Xe-133	0.02	0.04	0.05	0.08	0.11	0.12	0.12	0.12
Xe-135m	0.04	0.07	0.08	0.11	0.12	0.12	0.12	0.12
Xe-135m	0.06	0.12	0.16	0.26	0.34	0.38	0.39	0.39
	0.35	0.54	0.67	0.91	1.10	1.18	1.19	1.19

13.2.2 Insertion of Maximum Allowed Excess Reactivity

The analyses of this transient utilizes the reactor physics and reactivity coefficients determined by MCNP5 as described in Chapter 4, and thermal-hydraulic parameters determined by NATCON as described in Chapter 4, and the PARET/ANL[2] code. For this accident, the initial power before the transient was 12 kW.

The original PARET code has been adapted by the Reduced Enrichment for Research and Test Reactors (RERTR) Program to provide transient and thermal-hydraulics analysis for research and test reactors with both plate and pin-type fuel assemblies. The PARET/ANL version of the code has been subjected to extensive comparisons with the SPERT I and SPERT II (light and heavy water) experiments. These comparisons were quite favorable for a wide range of transients up to and including melting of the clad. Revisions of the code include new and more appropriate heat transfer, departure from nucleate boiling (DNB) and flow instability correlations, improved edits, reactor trips, control insertion model, a decay heat power model, and a loss of flow model.

The rapid insertion of the maximum worth of reactivity excess for PUR-1 (0.6% Δk/k) as specified by the Technical Specifications was evaluated. An assumption was made that the period trip (7s) failed, and scram was initiated on the power trip at 12 kW. Since the assumed measurement uncertainty on core power is 50%, a core power trip setting of 18 kW was utilized in the accident calculation. A delay of 0.1 seconds from the sending of scram signal to beginning of control rod motion is assumed. Fuel/coolant channels that are representative of the hottest fuel plates (as identified in Chapter 4) were modeled in the PARET analyses.

Results of this transient are summarized in Table 13-15. The reactor power increases from 12 kW to the trip setting of 18 kW in less than 2.5 seconds. There is a negligible increase in the clad temperature as a result of this hypothetical accident. The safety limit is never in danger of being reached.

Table 13-15: Peak power and clad temperature for insertions of 0.6% Δk/k with scram.

SCRAM	Reactivity Inserted	P0 (kW)	Pmax (kW)	Time of Peak Power (s)	Tclad,max (°C)		
					at t=0	at Peak Power	Max
YES	0.6% Δk/k step	12	46.4	0.173	57.40	57.40	57.40
YES	0.6% Δk/k over 10 s	12	18.4	6.25	57.40	47.75	57.45

These results demonstrate the ability of the LSSS to protect the safety limit of fuel temperatures not to exceed 530°C. The maximum temperatures achieved in the fuel are well below temperature of incipient boiling as well. Therefore PUR-1 can maintain the fuel integrity during this accident scenario.

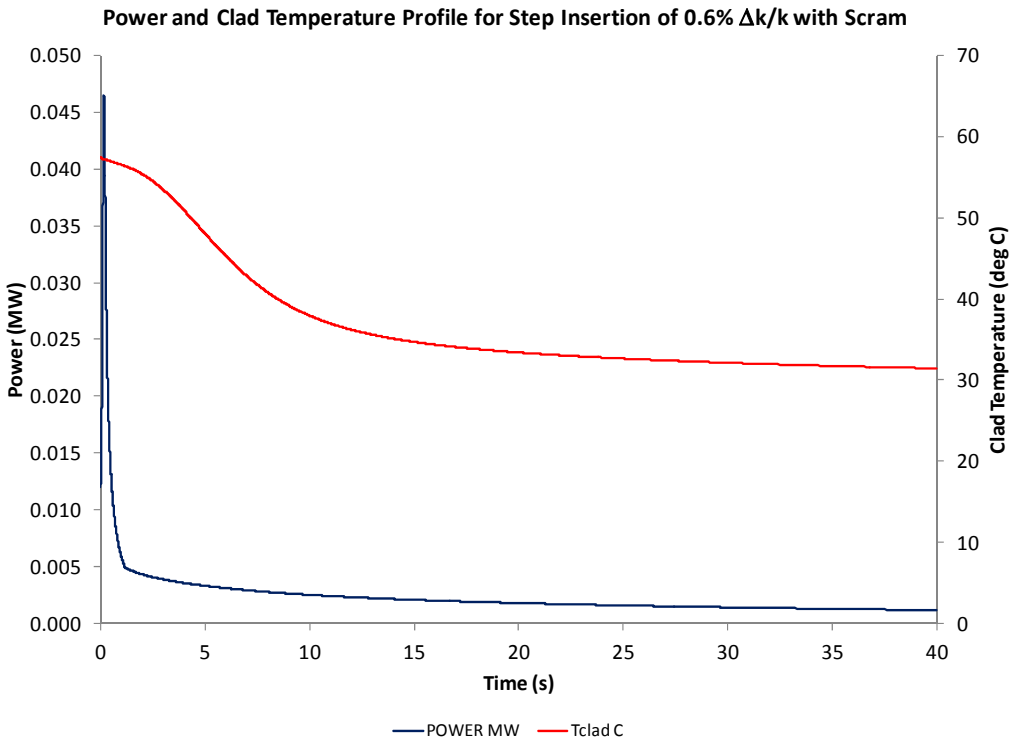

Figure 13-1: Power and Clad Temperatures for 0.6%Δk/k step insertion with scram.

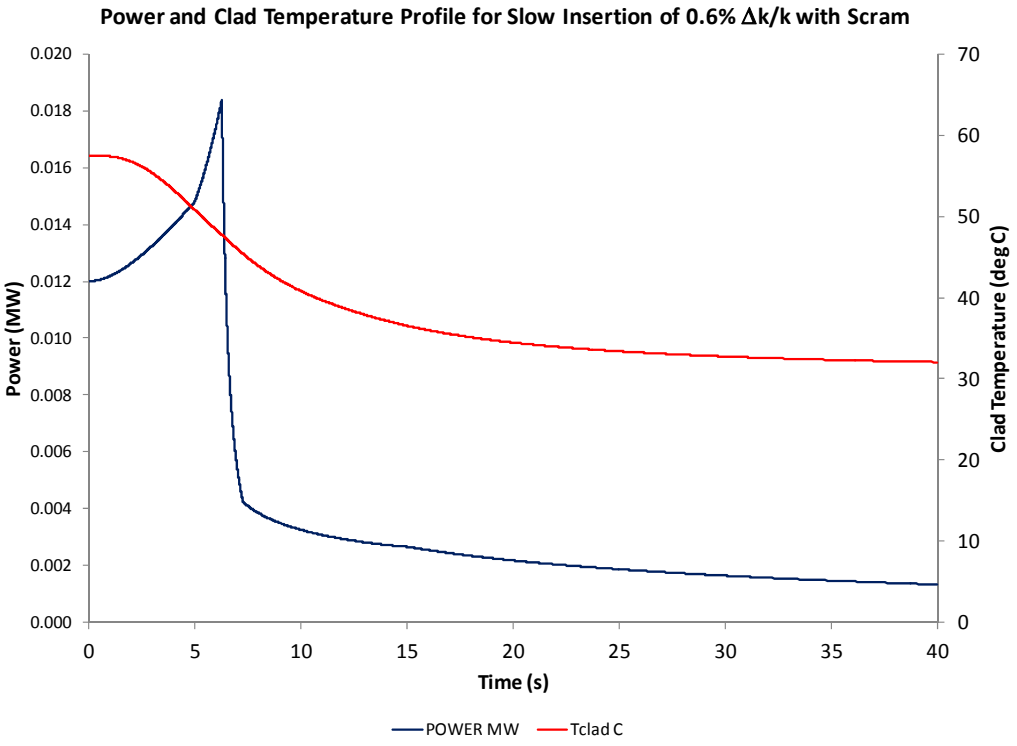

Figure 13-2:Power and Clad Temperatures for 0.6%Δk/k slow insertion with scram.

13.2.2.1. Insertion Without Scram

While it is an extremely unlikely event that all reactor protective systems will fail, the analysis of the step and slow insertion of the maximum excess reactivity without scram was analyzed. In both cases it is assumed that both shim-safety rods are completely removed from the core, and are not available for insertion to shut the reactor down. For this accident, the initial power before the transient was 10 kW.

The reactivity coefficients determined in Chapter 4 act in this case to keep the reactor power from increasing without bounds. The power in both cases (step and slow insertion) levels off at about 2.38 MW, with the maximum clad temperature reaching 133.08°C. This temperature is still well below the safety limit of 530°C, so there is no danger of failure of the clad. It should also be noted that emergency boration is available to the operators, and this would act to shut down the reactor in the unlikely case of failure of the reactor protection system. The peak power and clad temperatures are summarized in Table 13-16, and the power and clad temperature history for the accident are shown in Figure 13-3 and Figure 13-4

Table 13-16: Peak power and clad temperature for trip and no-trip insertions of 0.6% Δk/k.

SCRAM	Reactivity Inserted	P0 (kW)	Pmax (kW)	Time of Peak Power (s)	Tclad,max (°C)		
					at t=0	at Peak Power	Max
NO	0.6% Δk/k step	12	2591	674	54.43	133.56	133.56
NO	0.6% Δk/k over 10 s	12	2591	690	54.43	133.56	133.56

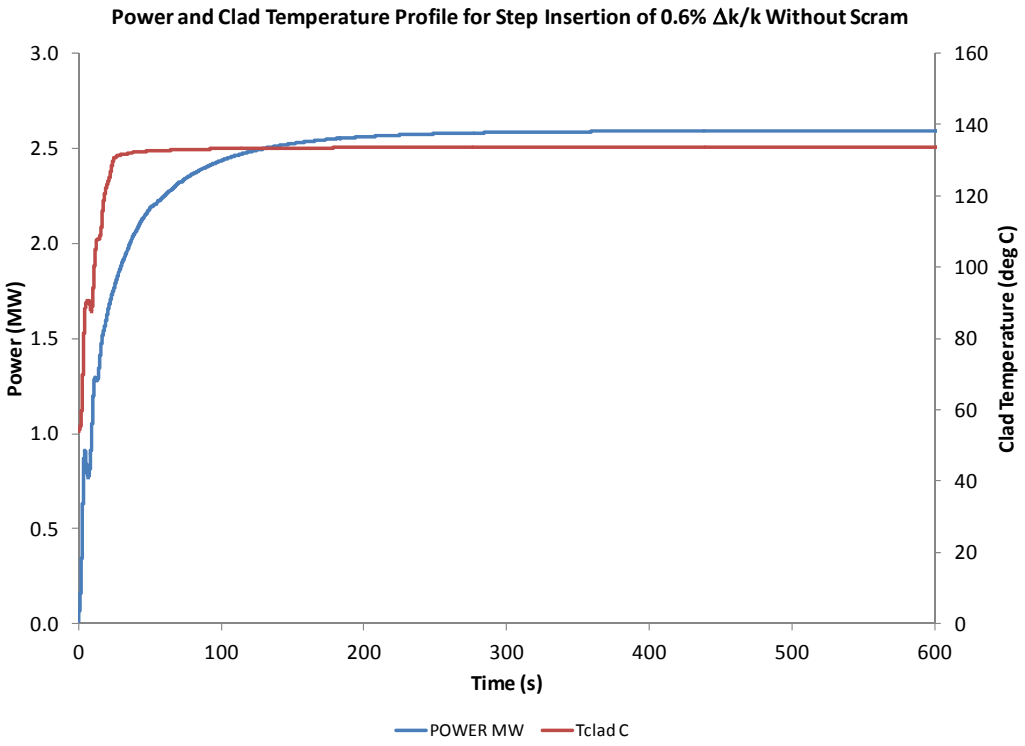

Figure 13-3: Power and Clad Temperatures for 0.6%Δk/k step insertion without scram.

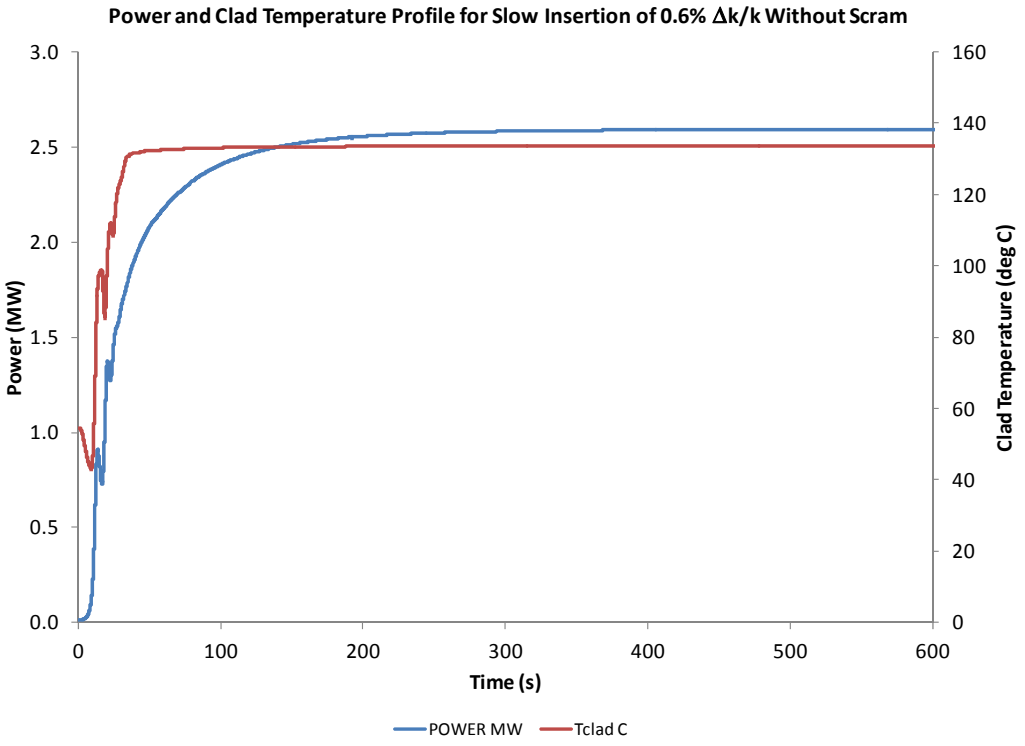

Figure 13-4: Power and Clad Temperatures for 0.6%Δk/k slow insertion without scram.

13.3 Summary and Conclusions

Evaluation of the accident scenarios detailed in this chapter lead to the conclusions that the PUR-1 reactor can operate safely and effectively at 10 kW continuously, with the reactor limiting safety system settings of 110% (11 kW) and 120% (12 kW) adequately protecting the reactor and its systems. Even in the very unlikely event of a complete failure of the reactor protection system and insertion of the maximum allowable reactivity, fuel integrity is maintained and there is little or no risk to the public.