NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
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U.S. Nuclear Regulatory Commission
Washington, DC 20555
Attn: Document Control Desk

Re: Response to Initial Request for Additional Information on the Relicensing of the MIT Research Reactor, License No. N-37, Docket #50-20.

Gentlemen:

In July 1999, the Massachusetts Institute of Technology submitted an application for the relicensing of the MIT Research Reactor. Three partial requests for additional information were subsequently received. We refer to those three requests here as the "initial" request for such information. Enclosed is our complete response. This material has been reviewed and approved by the MIT Reactor Safeguards Committee.

Some of the material contained in our response will require a change in the wording of either the safety analysis report and/or the technical specifications that were submitted as part of our July 1999 application which was designated as Revision 1. We will make those changes once we are certain that our response has effectively addressed all NRC questions. We are adopting this approach to avoid the creation of multiple drafts of very lengthy documents. It is our understanding that NRC concurs with this approach.

Sincerely,

Lin-Wen Hu, Ph.D.
Acting Director of Reactor Operations

John A. Bernard, Ph.D
Principal Research Engineer

cc: USNRC - Senior Project Manager
NRR/ONDD

USNRC - Region 1 – Project Scientist,
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FRSSB/DRSS

A020

1. The safety analyses were performed for [redacted] fuel form only. Discussions in the SAR related to the [redacted] fuel form will be deleted.

2. Section 2.2.1: CONRAIL on page 2-6 will be changed to CSX Transportation. CSX and Norfolk Southern Corporation announced the joint acquisition of CONRAIL in April 1997, and filed with the federal Surface Transportation Board (STB) in June of that year their joint application to acquire and allocate Conrail's predominantly Northeastern routes. The STB voted unanimously on June 8, 1998, to approve the transaction. CSX and Norfolk Southern began operating most CONRAIL lines and facilities on June 1, 1999, and the northeastern states of the United States are now run by CSX transportation.

   a) The former CONRAIL line, now CSX, is principally used for passenger train transfers and transportation of non-hazardous cargo, such as fruits and vegetables. Transportation of hazardous materials rarely occurs. The only material possibly considered hazardous that is occasionally transported is ethyl alcohol which occurs approximately once per month. Accordingly, derailment or other accident would not cause damage to the reactor. Should such occur, the situation would be managed by the City of Cambridge. City agencies periodically conduct exercises to test their capabilities to respond to incidents of this type. A derailment in the vicinity of the MIT Research Reactor is not considered credible because all trains are required to stop before crossing Massachusetts Avenue. Thus, train speeds when passing near the reactor are always very slow. A NUREG report titled, "Re-examination of Spent Fuel Shipment Risk Estimates," (NUREG/CR-6672, March 2000) analyzed train and truck accident scenarios for spent fuel shipment and dose rate estimates of cask failure. In the train accident analysis, the derailment probability is calculated to be 0.77% per train for non-collision derailment for a rail route of 1210 km. Therefore, the probability per km is 6.36×10^{-6}/km. (Note: CSXT reported a train accident rate of 1.92 per million train miles in 1996. That figure corresponds to 1.19×10^{-6}/km). Assuming the affected radius is 1 km, the derailment probability in the vicinity of the reactor exclusion area is 6.36×10^{-6} per train.)

   b) Ammonia Alarm: An Abnormal Operating Procedure (AOP) is currently in place for the ammonia alarm. It provides for a graded response that is a function of the detected concentration. The initial action is to notify the MIT Police. The ultimate action (80 ppm ammonia) is to shut down the reactor and isolate the containment building. (Note: As of this writing the building that contains the ammonia refrigerant is undergoing renovation and the ammonia has been removed. There are no plans to reinstall it. So, this issue may well become moot.)

3. Section 3.1.1.5:

a) The reactor containment was constructed in the 1950s. The criterion of the aluminum-water reaction was chosen at that time. It is mentioned in the new version of the SAR for historical reasons and in hindsight it would have been better to omit any mention. The paragraph in question will therefore be deleted. The aluminum-water reaction is neither a credible accident nor a credible source of pressure, and it is therefore not included in Chapter 13.

The following historical information may be of interest. The original MITR SAR bases its comments about the Al-H_2O reaction on a document entitled, "Final Hazards Summary Report to the ACRS on a Research Reactor for the Massachusetts Institute of Technology, MIT-5007 dated January 1956. We do not appear to have a copy of this document. Eight years later (1968), the then Director of the MITR, Dr. Theos J. Thompson, together with Dr. J. G. Beckerly, published the "Technology of Reactor Safety," MIT Press. Chapter 17 concerns chemical reactions and we quote:

> "The results of experimental studies suggest very strongly that the temperature reached by the aluminum metal primarily determines the nature of the reaction. Isothermal studies have shown that the oxide film is very protective up to 1300 °C (2372 °F). This is fully consistent with the repeated failure of many investigators to obtain a vigorous metal-water reaction by pouring molten aluminum into water. Only in the explosion dynamometer tests was the metal temperature high enough to produce a significant reaction.

A review of current reactor design and operating practices indicates that there is no reactor-induced mechanism for pressurizing the containment building.

b) A scenario that would require manual initiation of the pressure relief system is containment heatup from solar radiation when the ventilation system is inoperable and the intake and exhaust dampers are closed. Over-pressurization would result from expansion of the air that was trapped within the containment building. In the past, it was observed that the rate of pressurization is about 2 inches of water or 487.88 Pa per hour. Assuming that the heatup lasts for 12 hours, the overpressure would be:

$$\Delta P = 487.88 \text{ (Pa/hr)} \times 12 \text{ (hours)} = 5854.6 \text{ Pa or } 0.85 \text{ psia}$$

Using the ideal gas law and assuming that the initial temperature is 20° C, the temperature of the containment air that corresponds to that overpressure is

$$T = [(5854.6\ \text{Pa} + 10133.0\ \text{Pa})/10133.0\ \text{Pa}]\ (273\ \text{K} + 20\ \text{K})$$
$$T = 309.9\ \text{K} = 36.9\ \text{C}$$

The containment building is equipped with a pressure relief system that is intended to relieve pressure and not release any radioactivity that might be present. This system is equipped with a blower that is used for test purposes. In an actual overpressure situation, the building overpressure would be allowed to create airflow through the relief system and its filter bank (roughing, absolute, and charcoal). If radioactivity were involved such as in the MHA, then the pressure relief system would be placed on-line.

c) A scenario that could cause an increase in the vacuum in the containment building is that the intake damper closes (or intake fans shutdown) with the exhaust damper and fans on. The following interlocks would prevent a negative pressure that opens the vacuum breakers.

Interlock	Action
Damper – Fan Interlock	Stops intake and exhaust fan and auxiliary fans if any main or auxiliary damper closes.
Main Fan Sequence Interlock	Intake fan must be operating in order for exhaust fan to operate.

During the past 40 years of MITR operation, these interlocks have not failed and caused the vacuum breakers to open.

The normal intake and exhaust air flow rate is 4,000 cubic feet per minute. The total volume of the containment is 2,000,000 cubic feet. Using a conservative approximation that assumes a linear pressure reduction, the maximum decompression rate can be calculated as follows:

$$\frac{4000}{2{,}000{,}000} \times 14.7(\text{psia}) = 0.029\ \ (\text{psia/minute})$$

The design pressure at which the first set of breakers operates is between –0.015 and –0.036 psig. Therefore, the first set would open in about 31 to 74 seconds. The design pressure for the second set is between –0.036 and –0.062 psig. Therefore the second set would open in about 74 to 128 seconds. Note that this scenario is not considered to be a credible accident because of the redundant design of the interlocks.

4. Section 3.1.1.7: TS 3.7.2(a) provides for a dilution factor of 50,000. Its basis is given in the technical specification. TS 3.7.2(b) establishes corrective actions should the tritium concentration in the secondary coolant system exceed 1 μCi/liter. Its basis is also given in the technical specification. Please see our response to question 69 which has resulted in changes to TS 3.7.2. No exemptions to 10 CFR 20 regulations on effluents are being requested. Therefore, the last sentence of Section 3.1.1.7 will be changed to indicate this by deleting the words, "unless specific exemptions exist in the MITR Technical Specifications."

5. Section 3.1.2.5: The terms "fuel melting," "core damage," and "clad softening" were used interchangeably throughout the SAR and TS. The criterion for all is the clad ([redacted] The terms "fuel melting" and "core damage" will be replaced with "clad softening."

6. Section 3.1.2.6: Please see reply to question 5. The reactivity limit was established by determining when the [redacted] during the transient.

7. Section 3.1.2.7: No. The objective is to be able to monitor reactor power, core tank level, and coolant temperatures at all times (shut down and operating). To that end, these parameters are monitored by redundant and diverse means. Also, selected instruments are on emergency power or on battery.

8. Section 3.1.3.4: "Off-scale" means that the instrument is not capable of responding with the degree of accuracy specified in the cognizant calibration procedure. The period channels have high and low level trips that initiate the off-scale signal. The low level trip is set at a sufficiently low current to indicate an abnormal condition such as detector failure. Similarly, the high level tip is set at a sufficiently high level to indicate an abnormal condition such as saturation. Both of these trips occur before the period channels are actually non-functional.

9. Section 3.1.3.5: Cable runs were mentioned as an example. Information on instrument design is given in subsequent chapters of this SAR. See Ch. 5 and Ch.-8 in particular. Some of the approaches used at the MITR to achieve diversity include:

   - Nine nuclear channels: three essentially identical period channels, three identical power level channels, and three others for a power level scram, indication, and control. Thus, there are five designs in all.

   - Battery-operated power supplies are used for some instruments including nuclear channels.

   - Emergency power is used for many instruments.

- Coolant level and flow sensors (three each) are of different designs and use different power sources (electricity/compressed air).

- Primary temperature is monitored by a capillary sensor and a thermocouple, both of which provide diversity in temperature detection.

Common mode failures are avoided through the use of diverse designs, locations, and power supplies.

10. Section 3.1.5: The last sentence of this section will be changed to read, "The program achieves the objective of ANSI/ANS 15.17-1987." Also, the text of that section will be revised to add the following as a continuation to the last paragraph. "Specifically, that standard sets out fire protection objectives that: 1) there is reasonable assurance that safety-related systems can perform their required functions; and 2) there are defined loss criteria (limits for risk to personnel, radioactive or toxic contaminant release, etc.). The program components of the MITR plan achieve this by including such features as:

a) Passive:

- Separation of cables and instruments.

- Automatic reactor shutdown on interruption of electrical power and/or instrument signals.

- Minimization of combustible materials.

b) Active:

- Smoke detection system.

- Fire Suppression in hot cells.

- Manual fire control measures in the building.

- Use of fire watches/strict controls for cutting/welding.

- Aggressive housekeeping within the containment building to keep combustible and potentially volatile materials out of the building.

c) Assurance:

- Monthly area inspections.

- Calibration program for all suppression equipment and active components.

- Training of local fire-fighter responders."

11. Section 3.2.1: The reactor containment consists [redacted]. Hence, there will not be any damage.

12. a) Section 3.5: The [redacted] limit is specified in TS 5.3.1.

    b) Section 3.5(a)(iii): TS 5.3.1 will be modified to include the Uranium-235 loading of [redacted].

13. a) Section 3.5: The current specification for MITR fuel is [redacted]. We wish to change this to [redacted] in order to bring our fuel specification into agreement with DOE-INEEL recommendations. Specifically, we quote from a summary of ATR Fuel Plate Development:

    "From the beginning of development of [redacted] fuel program in the early 1960's, the ability of the fuel plate, i.e., cladding and core matrix, to retain fission products and fission gases was investigated extensively for all combinations of materials after irradiation. Core swelling of the fuel plate as a result of the volumetric changes due to fission product formation in the fuel matrix was calculated theoretically and measured on irradiated samples. Laboratory measurements were always found to be less than the calculated solid fission product swelling which predicted 6.38% increase for every $1x10^{21}$ fissions per cubic centimeter. It was determined through testing that fuel plate core swelling was dependent on three main variables: 1) core porosity or voids due to less than 100% density of the fuel core at fabrication, 2) burnup due to the formation of solid fission products and gases, and 3) temperature of irradiation where higher temperature resulted in sintering effects and an initial volume reduction followed by an increase.

    The attached figure from the Idaho Nuclear Corporation, Metallurgy and Materials Science Branch Annual Report Fiscal Year 1970, UC-25, page 45, shows test results of these three variables on fuel plate core swelling. Samples used in this test had void contents ranging from 2% to 16%. As

shown on the figures, an additional variable of fines or fuel particles less than the required +325 US Seive size was analyzed. [Note: The figure referenced above is in the Idaho report and is not reproduced here.]

Void volume limits of [redacted] fuel plates were subsequently established for the Advanced Test Reactor (ATR) Fuel Element Specification. This range was established with the knowledge that metallurgical fabrication methods normally produce [redacted] powder of [redacted] of theoretical density, which will always result in an initial void volume and was found to be beneficial in delaying or minimizing fuel plate swelling with burnup. The upper limit allows a high burnup (e.g., $2x10^{21}$ for the ATR fuel element) under all anticipated temperatures conditions with less than 7% core swelling which was considered acceptable for design life of the fuel element during the fuel development program."

The proposed increase in the percentage of [redacted] improves safety because more space is available for fission product gas accumulation. This is illustrated by the above referenced figure. In view of the above, the following changes will be made:

- Section 3.5 (a)(iv): Correct as is.
- TS 3.1.6.3: Peak fuel burnup shall not exceed $2.3x10^{21}$ fissions/cm^3 for fuel that is intermetallic [redacted]
- Section 4.2.1: Second sentence of last paragraph on p. 4-5 will read, "specifically, the [redacted]
- Table 4.2: Line B(2) will read, '[redacted]

b) Section 3.5 (a)(v): The alloy fuel option, [redacted] will be deleted from the SAR. Please see reply to question 1. Section 3.5.(a)(i) will be changed to say, ".[redacted] in the fuel matrix." Also, TS 3.1.6.3 will be modified to delete mention of UA1 fuel. (See revised language above.)

14. Section 4.1: The licensed power level on page 4-1 will be changed to 6.0 MW.

15. Section 4.2.1: Please see reply to question 1.

16. Table 4.2: Please see reply to question 1.

17. Section 4.2.1:

a) Incipient excess outgassing refers to an elevated fission product gas release that might be an indication of possible cladding deterioration. It might also mean something else such as the presence of tramp uranium. Normally the fission product gas is less than 1-3 % MPC (maximum permissible concentration). Higher values warrant investigation. Fuel 'sipping' would then be performed with the reactor shutdown to determine if any element's clad is deteriorating. Sipping entails collecting coolant that has been drawn through each fuel element and measuring it for fission products. If one element's clad is deteriorating, the collected coolant will have an elevated fission product concentration. We would then remove the element. In this way, outright clad failures (which would result in much higher fission product gas concentrations in the coolant as a whole) are precluded.

b) [redacted]

18. Section 4.2.2.1: A sample shutdown margin calculation is provided in Section 4.5.3.3. Please see p. 4-51. That calculation is for early in core life (criticality attained at 8.0 inches with one blade and the regulating rod stuck full out.) So, this example covers Parts (2) and (3) of the question. For Part (1), if a second blade were non-functional, the margin would be (2.53-2.26) or 0.27 beta, if that second blade were stuck full out (worst case). If it were at the bank height, the margin would be greater.

19. Section 4.2.2.5: There is no credible failure that could result in the maximum allowed ramp reactivity insertion rate. The calculated maximum allowed ramp reactivity insertion rate is 3.8 beta/minute (or 63 mbeta/s). As shown in section 13.2.2.2, the reactor would be shut down safely before the safety limits are exceeded if this ramp insertion rate occurred. However, it can't occur. The insertion rate for the maximum shim blade differential worth is about 11 mbeta/s (157 mbeta/inch at 4.25 inch/minute), which is much lower than the allowed limit. The shim blades are normally operated at a fixed speed. For some of the digital control experiments that were done in the 1980s, a variable speed motor was used on one blade. It allowed speeds up to a maximum of 150% of normal or ~6.4 inches/minutes. This corresponds to 16.5 mbeta/s, which is still well below the limit.

20. Section 4.3.1.1: The last sentence of the first paragraph is changed to read, "Thus, the design pressure (60 psig) is the pressure at which it exhibits ductile failure. The rated pressure (24 psig) is the steady-state operating pressure. The

working pressure (flow present) is 14.2 psig. So, the ratio of design to working pressure is 60/14.2 = 4.23. The rated pressure is higher than the normal operating pressure by 69%." (Note: 24 psig-14.2 psig)/14.2 psig = 0.69.)

21. Section 4.3.1.2: Na-24 is produced by an (n,α) reaction on aluminum. Its half-life is 14.96 hours. We consider the activity to be "significant" if the dose rate at one foot above the top of the coolant equals or exceeds 100 mR/h. The text, as written, is misleading because it implies that radiation protection procedures are observed only if the Na-24 activity is "significant." In reality, certain procedures are observed at all times with extra ones being in place if the Na-24 is significant. The text will be modified to read, "If the reactor is shutdown but Na-24 activity is still significant ($\geq$100 mR/h), then additional administrative procedures..."

22. Section 4.3.1.6: The polyethylene packing between the core shroud and the core tank was used as a gasket to form a tight seal. There is an Al6061 packing spacer ring underneath the polyethylene. The packing spacer ring would prevent bypass flow if the polyethylene gasket fails. The polyethylene packing and the packing spacer ring have not been replaced since the startup of MITR-II. Failure of the ring could result in an increase in bypass flow (i.e., flow that does not go through the core). This would be sensed by primary flow sensors MP-6 and MP-6A which measure the differential pressure (ΔP) across the core. Each instrument has a tap at the core inlet plenum and at the primary outlet flow point. These instruments are shown and described respectively in Fig. 5-1 and Section 5.2.1.2, p. 5-6 but neither the figure nor the write-up make clear the role of these two sensors in monitoring core ΔP. This section of the SAR will be revised to do so. The ΔP measured by these sensors is indicative of the coolant flow through the core and an automatic scram occurs if that flow is lost. See Table 3.2.3-1 of Technical Specification No. 3.2.3 and especially note (4) of that table.

    We estimate that the polyethylene gasket is exposed to radiation levels of approximately 1000 R/h. Even though the material may become embrittled earlier, 10^9 Rads are need for material degradation of the polyethylene. This would require about 100 years of continuous operation.

23. Section 4.3.1.6: The fuel storage ring is inspected periodically for structural integrity, currently quarterly. Corrosion is controlled by specifying both the conductivity and pH of the primary coolant. A bounding analysis for the fuel storage ring was performed in 1980 (SR#O-80-12) to calculate the K-effective under various conditions. That analysis concluded that, assuming all 29 positions were filled with fresh [redacted], the K-effective will not exceed the TS limit of 0.9 in the absence of the cadmium. For the case of 29 fresh [redacted] the calculated K-effectives are 0.465, 0.448, and 0.825, respectively. Therefore, although the cadmium absorber is required by TS 5.4.3 (b), it is not needed to satisfy TS 5.4.4.

24. Section 4.4.3: A dedicated cleanup system maintains the quality of spent fuel pool water and thereby controls the corrosion of the spent fuel storage rack. The conductivity and pH of the water is sampled periodically to ensure quality. A bounding analysis for fuel storage in the spent fuel pool was performed in 1980 (SR#O-80-13) for four spent fuel storage racks filled with fresh [redacted] of fuel elements, as if they formed a large 20 by 20 rack. The calculated k-effective is 0.73 for racks with [redacted]. This is lower than the TS limit of 0.9. Note that fresh fuel elements are not allowed to be stored in the spent fuel pool and therefore this analysis is a very conservative one. A procedure similar to the one for inspection of the fuel storage ring in the core tank is used for the spent fuel pool. Its frequency is also quarterly.

25. Section 4.5.1.9: The statements refer to two different things. A model of the MITR core has been developed for MCNP and that model's results have been benchmarked and published. Hence, MCNP can be reliably used for flux calculations and for thermal-hydraulic calculations based on those flux values. A fuel management program that uses MCNP is under development. Statements in sections 4.5(b) and 4.5.2.3 refer to the fuel management program. The other statements refer to the MITR model and its use with MCNP.

26. Section 4.5.3.4: Non-fueled in-core positions are filled with either a solid aluminum dummy element or an in-core facility, such as the ICSA or an in-core loop. ICSAs and in-core loops have their dedicated dummy elements to prevent bypass flow through the gaps between the loops and the adjacent elements. The effects of solid dummy elements and ICSAs were examined during the MITR-II startup testing when measurements were made of the various flow factors. Every element position was measured for core configurations of one to five dummies. These measurements were repeated several years later and did confirm them. Tables of the flow factors are available and are used in the fuel management program to evaluate the safety and operating limits for the core in question. If a new core configuration (i.e., one for which a table does not already exist) is planned, the measurements will be redone. Please refer to T.S. 3.1.4.4, especially parts c and d.

27. Section 4.6.2.2:

    a) References for equation (4-5) should be [4-14, 18, 19] instead of [4-17]. The channel outlet subcooling ratio, R, is shown as Q_{ratio} in references [4-18] and [4-19].

    b) The equivalent diameter De in the SAR is defined as $D_e = 4A_f / P_w$ where A_f is the flow area and P_w is the wetted perimeter of the coolant channel. This is the standard definition used in text books [ref. 4-16]. The flow area between two fuel plates is $A_f = 1.2490 \times 10^{-4}$ (m^2) and the wetted perimeter is $P_w = 0.2285$(m). The heated equivalent diameter is defined in reference [4-14] as $D_h = 4A_f / P_h$ where A_f is the flow area and P_h is the heated

perimeter of the coolant channel. The difference between P_h and P_w is the non-heated gaps on the sides which is about 5 mm for both. Therefore, P_h= 0.2285-0.005=0.2225 (m). Throughout the calculation of OFI, for simplicity, it is assumed that $D_h = D_e$ because the difference is very small (less than 3%). Furthermore, using D_e in Eq. (4-8) yields a more conservative result in the OFI calculation.

28. Section 4.6.6.1: Coolant density at 50 C; ρ=988 kg/m^3, Flow area between two fuel plates: [redacted]. Therefore, mass flow rate in kg/s is

1800 (gpm) x 63.09E-6 (m^3/s/gpm) x ρ(kg/m^3)=112.2 (kg/s)=$\dot{m}$

Average primary flow through a coolant channel, using [redacted] in core and a coolant flow factor (F_f) of 0.921, is

[redacted]=2298.2 (kg/m^{2}s)

Apply the worst-case channel flow disparity factor of 0.864, as shown in Eq. (4-18):

$$2298.2 \times 0.864 = 1985.6 \cong 2000 \text{ (kg/m}^2\text{s)}$$

29. Section 4.6.6.2: The first part to this answer is the reply that was originally prepared. The second and third parts are in reply to additional questions that arose when we requested clarification of the original question. (Note: A file memo dated 10/14/03 summarizes the fin treatment approach.)

a) Flow area between two fuel plates: [redacted] wetted perimeter P_w=0.2285(m). Calculate the equivalent diameter of coolant channel using

[redacted]

[redacted]

b) Please refer to Figure 7 of Reference 4-14 which is the paper by Whittle and Forgan that shows the quantity R as a function of L_h/D_h. Relative to that figure, we quote from the paper (p. 97, left column, 2nd full paragraph), "It appears that R falls rapidly with the L_h/D_h ratio when the latter is less than about 100, and rises slowly with L_h/D_h for values over 100." Thus, we conclude that the value of R is not sensitive to L_h/D_h once

L_h/D_h exceeds about 150. This justifies the use of equation 4-8 for an L_h/D_h of 260. A second argument for this conclusion is given in Table 4-6 of the SAR. That table compares the mass flow rates at OFI for the three available correlations and for the MULCH Code which solves the momentum/pressure drop equations for multi-channels. The agreement between MULCH and Equation 4-8 is quite good.

c) Treatment of the fins by use of a wetted perimeter is the standard engineering approach for such geometries. We believe that it is an acceptable approach.

30. Section 4.6.6.3

a) The reference citation for equation (4-30) should be (4-20), not (4-15).

b) A MCHFR (minimum critical heat flux ratio) of 1.5 is now applied to the safety limit for natural convection operation. The dry-out condition becomes 399/1.5=266(kW). A reactor power of 250 kW is adopted as the safety limit.

31. Section 4.6.7: There was a factor of 0.95 applied to the wall superheat $(\Delta T_{sat})_{ONB}$ that should be stated on page 4-73. This factor is used to ensure a lower bound of wall superheat although the correlation is said to yield a low incipient heat flux. Derivation of Eq.(4-34) is as follows:

$$T_{clad,ONB} = T_{sat} + 0.556 \times \left[\frac{q''}{1082P^{1.156}}\right]^{0.463P^{0.0234}} \qquad \text{Eq.(4-4)}$$

$$P = 1.3 \text{ bar}, \; T_{sat} = 107 \text{ (C)}$$

$$(\Delta T_{sat})_{ONB} = T_{clad,ONB} - T_{sat} = 0.95 \times 0.556 \times \left[\frac{q''}{1082 \times 1.3^{1.156}}\right]^{0.463 \times 1.3^{0.0234}}$$

$$(\Delta T_{sat})_{ONB} = 0.0177 \times \left(q''\right)^{0.466}$$

$$T_{clad,ONB} = T_{sat} + (\Delta T_{sat})_{ONB} = 107 + 0.0177(q'')^{0.466} \qquad \text{Eq.(4-34)}$$

32. Section 4.6.7.2: The natural convection *calculation* was performed as a transient with the reactor modeled as if it were operating under a forced flow condition at a high power level prior to the loss of forced flow. In the model, natural convection is then established at a constant reactor power. This is a strategy to obtain a numerical solution with less simulation time and is not a real mode of natural convection operation. The 1 MW initial power was chosen arbitrarily and is conservative because the decay heat load will be higher than the real operating mode which would be a startup under natural circulation conditions to 100 kW. A higher initial power would not affect the solution but would require a longer

simulation to obtain the natural convection solution. So, to summarize, the initial power level has no effect on natural circulation in the MITR. If the reactor loses forced flow, it shuts down. The natural circulation mode is never initiated from a high power level that requires forced flow.

33. Section 4.6.7.1:

a) The approach adopted by MIT to determine the LSSS and the SL is the same as that used in the SAR for the existing reactor that was first operated in the mid-1970s. That approach was to derive LSSS and SL values that are based on physical phenomena. These are onset of nucleate boiling (ONB) for the LSSS and either critical heat flux (CHF) or onset of flow instability (OFI), whichever occurs first, for the SL. See Section 4.6.5 of the SAR. This approach was reviewed and approved in the early 1970s by the MIT Reactor Safeguards Committee which, at the time, included some of the leading nuclear engineers (i.e., Rasmussen, Lanning, Driscoll, and Griffith) in the United States. The advantages to the approach are that: 1) it defines LSSS and SL in terms of actual physical phenomena that are relevant to the thermal hydraulic behavior of the core; 2) it provides a clear basis for calculation of the various limits, and 3) it is conservative in that the margin between the LSSS and SL is greater than if one only allowed for measurement and process uncertainties. We illustrate the last of these advantages below.

The question specifically asks about process uncertainty, measurement uncertainty, instrument response, and rod drop times. Process uncertainty is addressed by selecting initial conditions that correspond to the LSSS instead of the normal operational values or even the scram values. This was not done originally for all safety analyses but, as a result of other questions in this series, it has now been done. Measurement uncertainty is documented in Table 4-8 of the SAR for power (5%) and flow (5%). For primary temperature, it is 2 °C per MITR calibration acceptance tests and for primary level it is 1 inch, again from calibration acceptance criteria. Instrument response and rod drop times are listed in the individual analyses given in the SAR or in the answers to other questions in the series. Consider loss of primary flow as an example. An instrument time delay of one second is assumed and another second is assumed for 80% blade insertion. The maximum calculated fuel temperature of the hot channel is 132 °C based on the initial conditions of LSSS (T=60 °C, W=1800 gpm, P=7.4 MW, and L=10 ft.). Using the instrument uncertainties listed above the maximum calculated fuel temperature would be less than 135 °C, which is much lower than the SL criterion of fuel integrity, or the [redacted]. Note that the 1 inch uncertainty on level corresponds to less than 0.1 °C on coolant saturation temperature. Hence, the margin between the LSSS and SL is substantial.

The analysis is similar for other transients except that of the step reactivity insertion. There, an instrument delay time of 0.1 s and a linear negative reactivity insertion rate based on blade drop time for the step reactivity insertion calculation was used.

b) The safety limit is a function of four measurable parameters: core power, coolant flow, outlet temperature, and coolant level. A combination of these parameters is safe provided that : 1) The safety limit factor, as defined in Section 4.6.6.2 of the SAR, is less than 2.4; and 2) the particular combination of power, flow, and temperature is below the selected curve in Figure 4-25 of the SAR.

The safety limit factor is evaluated for each core configuration and documented as part of the refueling paperwork. Combinations of power, flow, temperature, and level are evaluated by reference to Figure 4-25. The development of this figure is as follows: Figures are two-dimensional and there are four parameters. So, we needed to reduce the number of variables. Power, flow, and temperature can vary continuously. In contrast, level can take on only certain stable values. One is with the coolant 4" below overflow which corresponds to a height of ten feet above the top of the fuel plates. Another is at the height of the anti-siphon valves which is six feet above the top of the fuel plates. So, one simplification for the creation of a figure was to preselect certain levels. A second simplification was to use the power-to-flow ratio thereby combining two variables into one. The result is Figure 4-25. The figure should not be interpreted as meaning that a coolant height of ten feet is the safety limit on level. Such a statement has no meaning because the SL is a function of all four parameters. Figure 4-25 should be interpreted as meaning that "for a coolant level of ten feet, any combination of power, flow, and outlet temperature that is below the ten feet coolant height curve will satisfy the safety limit."

c) The criterion for natural circulation SL is a zero flow condition where the core can be cooled by countercurrent flow with a downward coolant flow and an upward flow of bubbles or steam. The coolant temperature is not relevant as long as coolant inventory is maintained above the top of the fuel and the heat flux is below the calculated limit. Therefore, one could specify a SL of 100 °C, which is the boiling point neglecting any pressure from coolant above the core. However, because the coolant condition at this point would be two phase, temperature would always be at the saturation temperature and hence not vary. Also, it is not possible to locate a sensor this close to the core, and hence this temperature would not be measurable. For these reasons, we did not list it.

34. Section 4.2.1:

a) The corrosion film (crud) thickness is assumed to be 2 mil in the calculations of the SL and LSSS.

b) The thermal conductivity of the corrosion film is 2.08 J/mK.

c) No. Elements that approach the existing fission density limit are returned to DOE as spent fuel and hence are not available for reuse at the MITR.

d) There should be no increase in the percentage of elements removed for excess outgassing or other failure type. In our experience, excess outgassing only occurs early in life.

35. Section 5.1: The phrase "multiple failures" is a typo. The sentence should read "...in the event of a fuel element clad failure."

36. Section 5.2.1.3: The bypass flow of primary coolant was verified during MITR-II startup testing. It remains the same because the core configuration remains the same.

37. Section 5.2.1.11: Please refer to T.S. 3.3.6 which specifies a short interval as 48 hours and establishes quantitative limits (pH between 5.0 and 8.0, chloride ion is below 6 ppm).

38. Section 5.2.2.8: The low level alarm set point of the primary storage tank is 2" below the level at startup. It corresponds to about a 57 gallon decrease of primary coolant volume. (Note: This figure depends on the initial level because the tank is curved. The figure cited is the maximum.)

39. Section 5.3.2.8: The low level alarm set point of the dump tank is 2" below the level at startup. It corresponds to an 8 gallon decrease of reflector volume.

40. Section 5.4.2.4: Thermal conductance probe PC-1 reads out in the shielded portion of the equipment room. It can be accessed during reactor operation.

41. Section 5.5.1.5:

a) Please see the memo contained in Appendix A to this response.

b) The secondary coolant is pumped from the heat exchangers to the cooling tower and then returned to the heat exchangers. If radioactive material were present in the secondary coolant, it could be released to the environment from the cooling tower.

c) Please refer to the memo referenced above for the answer to part (a) of this question. The conclusion of that memo is:

"In summary, using very conservative assumptions it can be seen that: for undetected leaks, the maximum H-3 activity released over one year via either blowdown or evaporation will fall within Technical Specification limits; and for higher rate short-term releases, although the activity discharged would exceed 5 Ci, the maximum dose to a member of the public would be less than 20 mrem.

For Na-24, the maximum concentrations in an undetected leak occurring over one year will be less than 10 CFR 20 limits; and for higher rate short-term releases, the maximum dose to a member of the public would only be about 1 mrem."

42. Section 6.2: There is no conflict. The statements refer to two different things. The three hour cooling period after reactor shutdown in Section 5.2.4 is intended to make sure that the primary coolant is cooled to a low enough temperature so a large heat capacity is available for decay heat removal during a routine shutdown period, e.g., a scheduled outage. The discussion related to natural convection valves in Section 6.2 is for the scenario of a loss of flow accident.

43. Section 6.4: There is a typographical error in the second line of paragraph b at the bottom of p. 65. The ANSI/ANS standard to which reference is made should be Number 5.1 and not 15.1. This correction will be made.

44. Section 6.5.3.1: The aluminum window has been secured and has remained in the closed position since at least 1975. There is no plan to use the aluminum window for experiments in the near future. The window is covered on both its interior and exterior by bolted flanges. The flange seals are tested during every containment building pressure test. The seals have never shown any detectable leakage.

45. Section 7.2.1: Please see reply to question 9. In addition, the following is noted. The reactor protection system (RPS) is both redundant and diverse. The RPS has three period and three level channels for redundancy. Diversity is achieved by use of separate locations (some in instrument ports which are horizontal and below the core and some in GV ports which are vertical and opposite the core), separate cable routings, and use of normal and emergency power supplies.

46. Section 7.3.2.2: The proximity switches provide direct indication of the blade/rod positions.

47. Section 7.3.2.3: The blade-in lights indicate full insertion of the shim blade. However, they do not cause full insertion. There are two possible causes: loss of electrical power to the blade magnets that results in the blades dropping into the core (i.e., a scram or power outage) or a controlled insertion by means of the

blade drives. For both cases, it is desirable that both the blade and its associated drive be fully inserted. Hence, on loss of power to a magnet, the system is configured to run the drive mechanisms in even though the blades have already dropped to the full-in position. This action has the added advantage of providing further assurance that the blade is fully inserted. We don't feel that the sentence is confusing. We could delete the last eight words, "thereby ....," Or modify it to read, "thereby further assuring..." We recommend leaving the sentence as is.

48. Section 7.3.2.4: Yes. The scram results from an interlock on the dump valve itself. If the valve is not fully closed, the reactor will scram.

49. Section 7.4.2: The first line of the sentence in paragraph g should read, "DL-6 is an insulated stainless steel conductance probe that..."

50. Section 7.6.2: This sentence refers to the standard procedure of calibrating nuclear instruments against thermal power by means of a calorimetric measurement. Specifically, "equilibrium thermal power" refers to the calorimetric result. "Channel 8 equilibrium value" means the instrument reading during steady-state conditions.

51. Section 7.6.3: The nuclear instruments to which reference is made are those used for indication by the console operator and/or analog automatic control. The acceptance criterion is 5% although deviations are corrected before that limit is attained. (Note: Deviations are the normal result of fuel burnup and other things that alter the neutron flux shape.) The nuclear instruments that are used for the safety system are calibrated against thermal power prior to startup if shutdown were more than 16 hours and at least monthly.

52. Section 7.7: Not applicable. See response to question 4 and question 69.

53. Section 7.7.1.2: All area monitors are independent in that an alarm or failure of one unit does not disable another. All alarm locally and on a panel in the reactor control room. In addition, all units feed a common annunciator alarm on the reactor scam panel. Once any given unit alarms or is disabled, the annunciator alarm activates. Hence, if a second unit alarms locally and at the panel, it will not trigger the annunciator alarm because that alarm is already on. This is why we have the option to install bypasses on a unit that is de-energized or removed for maintenance. (Note: The control room monitor panel is to the immediate left of the reactor operator when he or she is seated at the console. The individual panel alarms are therefore clearly visible.)

54. Section 9.1.3: All interlocks listed in Table 9-1 are tested for operability at least annually. All such tests are specified by written procedure with results documented in our test and calibration files. All such tests are listed in our master schedule for test and calibrations. Many of these interlocks exist for reasons of

industrial safety as opposed to reactor safety, and hence are not listed in the technical specifications.

55. Section 9.2.2.1: The cleanup pump takes suction from a point approximately five feet below the pool's surface. If a pipe break were to occur in the spent fuel pool cleanup system, only about 25% of the pool's total volume could be syphoned out and spent fuel elements would remain adequately shielded.

56. Section 9.3.3: There are no interlocks between the smoke alarm systems and the containment building ventilation system. Moreover, none is desired. The building volume is so large that a shutdown of ventilation will not impact a fire by curtailing the air supply. Such an action would only result in the buildup of smoke to levels that might cause harm during the course of evacuating personnel. Once all personnel are accounted for, the ventilation can be shutdown from outside the building. (Note: All building exhaust air is monitored for radioactivity and if abnormal levels are detected, the containment building will be automatically isolated. Hence, if radioactivity is associated with any smoke, it will not leave the building.)

57. Section 9.4: Two-way radios were not practical for use within the containment building because the structure acts as a Gaussian surface and precludes signal transmission. We have recently installed a 900 MHz repeater system that allows the use of such radios by police/fire and others who utilize that frequency.

58. Section 9.5.1: Byproduct material produced by the reactor is transferred to the MIT Byproduct License (or to the Byproduct License of another organization) when the material leaves the reactor's restricted area. All such transfers are documented in writing with copies of the license on file at the reactor. If byproduct material produced by the reactor is to be used by MIT researchers (or others) within the reactor containment building or the associated restricted area, then it would remain on the reactor license.

59. Section 7.7.2.2: For purposes of comparison, both the LSSS and the SL should be computed using the same conditions. The last three lines of the second paragraph in Section 7.2.2.2 will be revised to read: "set at 6.6 MW, the Limiting Safety System Setting is 7.4 MW, and the Safety Limit is 9.1 MW. The LSSS and SL are calculated for the following conditions: coolant height, 10 feet; primary flow rate, 1800 gpm; and reactor coolant outlet temperature, 60 C."

60. Section 10.2.4.1:

a) No. Samples that are to be transferred from either a 1PH or the 2PH1 facility to NW13 via the carrier tube are first checked for radiation levels. If the level exceeds a specified threshold, the transfer is precluded. The threshold is set so as to preclude unacceptable dose during transfer and subsequent handling. The check is done via automatic interlock for 1PH1

samples (the principal tube) and manually for the others. In our experience, we have never had a sample become stuck in the carrier tube.

b) No. Samples transferred to NW 13, either manually or by carrier tube, are transferred to the MIT Byproduct License before being transferred. As noted in question 58 above, all such transfers are documented in writing.

61. Section 10.3.1: The MITRSC may specify if a previous experiment falls in one of the existing envelopes. If not, the division of authority is: 1) for in-core samples/experiments: the Director of Reactor Operations and the Reactor Radiation Protection Officer; 2) for ex-core samples/experiments: the Reactor Superintendent and the Reactor Radiation Protection Officer. Hence, in each case, two approvals are needed and those approvals are made independently.

62. Section 10.3.2: Yes. We have adopted the new format. The three criteria listed in this section will be replaced with the eight now listed in 50.59 (c)(2).

63. Section 10.3.2.6:

a) There are no weight limits on ICSAs because they are not supported by the upper and lower core grid plates. ICSAs are either supported by a support bridge that is placed in the core tank above the coolant level, or by the reactor top lid, if they extend through the plugs in the top lid.

b) ICSA thimbles fit into dedicated dummy elements. The exterior of the dummy element conforms to the dimensions of a fuel element. The clearance between the exterior of an ICSA thimble and the interior of its dedicated dummy is kept as small as possible. For example, for the IASCC loop, which is the most recent completed experiment, a standard in-core dummy will give a tolerance of 0.0175 inches on all sides of the thimble if perfectly centered. The amount of bypass flow is negligible because of the high flow resistance.

64. Section 10.3.2.8:

a) Digital control experiments are subject to the same review process as any other experiment. Specifically, a safety review and Q/A package are prepared. These are reviewed by the Reactor Operations Staff and then by the MITRSC.

(i) For Hardware: Written procedures are prepared and approved to ensure that hardware functions as intended. Surveillance frequencies are typically prior to the initiation of a digital control experiment session and at least monthly.

(ii) For Software: All software is tested via simulation. Also, check circuits (subroutines execute a certain sequence, time-outs that would trip if there were an infinite loop) are used.

b) The responsibility lies with both the experimenter and with Reactor Operations Staff as it does for any type of experiment. The experimenter is responsible for designing the system to the satisfaction of the Reactor Operation Staff who then independently verify its "as-built" behavior. All testing is done by Reactor Operations and all digital control experiments are observed by a licensed SRO who is not one of the experimenters.

65. Section 11.1.1.4: A paragraph will be added at the end of Section 11.1.1.4 which concludes that the maximum anticipated annual dose to workers from airborne radiation sources is calculated to be approximately 2.75 mrem/y [11-3]. This assumes a stay-time of 1000 hours/y by a watchstander in the control room. Also, a new reference will be added at the end of Chapter 11. It will be "11-3, File Memo (Dose Assessment from Airborne Activity Within Contained Spaces of MITR-II)." A copy of this file memo is provided in Appendix B.

66. Section 11.2.2.3:

a) Please see Appendix C to this response for a discussion of the calculations that form the basis of the dilution factor and dose scaling factor.

b) The maximum exposed member of the public would be an individual living in an MIT-controlled property that is ~100 m from the MITR. The dose at 6 MW from routine annual release of effluent to that individual would be ~1.2 mrem/year.

c) The nearest non-MIT residence is at approximately 250 m. The dose would be less than 1 mrem/year at 6 MW from routine annual release of effluent.

67. Section 11.2.2.4: We have implemented a new written procedure for all such discharges. The liquid that is to be discharged is recirculated (no discharge) while being filtered by a series of progressively finer filters. When sampling indicates that no particles in excess of 0.35 micro-m are present, a filter of that fineness is installed and the discharge is made. 10 CFR 20.2003, "Disposal by Release into the Sanitary Sewerage," refers to former ASTM standard D1888-78, which specified the use of a membrane filter that will remove all particles greater than 0.45 micro-m in size. This filter size, originally intended for testing purposes, has become the effective standard for sanitary sewer discharge filter systems because this presumably assures solubility during discharge without the need for filter solubility analysis on each discharge. The 0.35 micro-m filter that we are using is therefore conservative.

We have also implemented a program to minimize discharges of non-radioactive water to the waste tanks.

68. Section 11.2.2.6: No, none of it goes to the waste tanks. In reality, there have been no discharges of radioactive material via most of these sinks for many years. One sink is active and it is used to dispose of tritium sampling water and to wash sampling equipment. The following protocol is in effect for all sinks, whether in use or not:

- Certain sinks are designated as approved for discharges.
- Material must be soluble.
- An inventory is kept of any discharge and the records are tallied against the limits of the MIT Reactor License.
- Wastes that do not meet the discharge criteria are collected and solidified for disposal as low level solid waste.

69. Section 11.2.2.7: The 1 μCi/liter tritium concentration limit is intended as a limiting condition for operation that will ensure compliance with 10 CFR 20. Specifically, facility actions such as cessation of secondary coolant discharge and the securing of cooling tower spray are required upon detection of tritium at this level. These actions will minimize the total activity discharged to the sanitary sewer. This limit for the secondary system is 10 times more restrictive than the values specified within 10 CFR 20 Appendix B values for the monthly average concentration (MAC) for tritium (1E-2 μCi/ml). This level is also sufficiently greater than minimum detectable so as to constitute a practical measurement.

The annual blowdown rate from the cooling tower is 2E10 ml/y. Hence, if the tritium concentration were maintained at the LCO for the entire year, the total activity that would be discharged would be about 20 Ci. However, cooling tower water is required to be sampled daily (24 hour interval) for tritium. Hence, corrective facility actions would be implemented before a discharge in excess of 10 CFR 20 could occur.

The language in the first paragraph of Section 11.2.2.7 and in TS 3.7.2(b) is a now outdated holdover from the MITR-II SAR. This language will be changed to make clear that an exemption from 10 CFR 20 limits is not being sought. The new wording will be:

a) First Paragraph of Section 11.2.2.7:
Liquid waste is discharged to the municipal sanitary sewer systems from two waste storage tanks and from the cooling tower basin. Radionuclide concentration limits set on the monitoring and sampling systems are such that conformity with the limitations specified in 10 CFR 20 is assured.

b) TS 3.78.2(b):
On indication of 1 μCi/liter of tritium in the secondary coolant water, the cooling tower spray should be shut down, the secondary system water discharge shall be stopped, and the D2O reflector heat exchangers shall be isolated until tritium leakage into the secondary has been controlled.

c) Third Paragraph of Basis of TS 3.7.2(b): Same wording as first paragraph of Section 11.2.2.7.

70. Section 12.1.1: As noted in Section 12.1.1, MIT has two committees (Committee on Use of Humans as Experimental Subjects (COUHES) and Committee on Radiation Exposure to Human Subjects (COREHS) that address this issue. Also, our medical partners (a hospital) have similar committees. The protocols for the BNCT Program are reviewed and approved by these committees. (Note: Additional information of this nature was provided to NRC when MIT requested approval for use of its fission converter beam for BNCT. NRC has since approved that beam and, in doing so, reviewed and approved the entire BNCT program for regulatory conformance.)

71. Section 12.1.3: The question contains a typo. The senior reactor operator is required to be present in the control room (not merely at the facility) for these evolutions. This is a far more stringent requirement. A significant reduction "in power" is defined by written procedure as 10% or greater of the previous steady-state power.

72. Section 12.2.3 et al: The following changes will be made:

- Section 12.2.3(a) will be revised to read, "Determinations that changes in equipment, systems, tests, experiments, or procedures described in the annual report do or do not meet the criteria of 10 CFR 50.59 (c)(1).

- Section 12.3.2(d), will be revised to read, "a determination if the criteria of 10 CFR 50.59 (c)(1) are met."

- Section 12.3.2, last paragraph will be revised to read, "Procedural changes that would affect the basis of a technical specification or which otherwise do not meet the criteria of 10 CFR 50.59 (c)(1) require approval of the U.S. Nuclear Regulatory Commission."

- Section 12.9.1(c) will be revised to read, "verification that the criteria of 10 CFR 50.59(c)(1) are met."

- TS 7.2.2(a) will be revised to read, "Determinations that changes in equipment, systems, tests, experiments, or procedures described in the

annual report pursuant to Specifications 7.7.1.4 and 7.7.15 do or do not meet the criteria of 10 CFR 50.59 (c)(1)."

- TS 7.4.1.3(a) will be revised to read, "the criteria of 10 CFR 50.59(c)(1) are not met, or"

- TS 7.4.2, last paragraph, will be revised to read, "In the event that the review required by Specification 7.4.1 identifies a situation in which the criteria of 10 CFR 50.59(c)(1) are not met, or finds that the emergency or security plan is degraded or that the requalification program is weakened or that the ALARA program is negatively impacted, then the proposal must be referred to the MITRSC (and possibly to the NRC)."

73. Section 12.3.1: There is an inadvertent omission from the list on p. 12-15. The list should have included "Administrative Procedures" that include administrative objectives and requirements concerning both the Radiation Protection Program and the shipment of radioactive materials. In addition, the MIT Reactor Radiation Protection Office prepares and maintains procedures for implementing the Radiation Protection Program (i.e., survey and sample methods, instrument calibrations, surveillance frequencies, etc.)

74. Section 12.3.3: The evaluation of temporary changes is made by the approving individuals, the minimum of which would be described in 12.3.3 (b) and (c). As part of this evaluation, the criteria listed in 10 CFR 50.59(c)(1) are followed. (Note: Procedures are listed in the Reactor Administrative Procedures as to whether or not MITRSC approval is required. Hence, the individuals making the changes here do not also determine if MITRSC approval is required.)

75. Section 12.3.3: This question refers to 12.2.2, but we believe 12.3.3. is meant. Reference to 10 CFR 50.72 will be deleted from paragraph (d).

76. Section 12.5.1: There are four parts to the reporting requirement. These are: 1) total radioactivity excluding tritium; 2) total tritium; 3) identity of specific nuclides reported under (1) if the activity exceeds $1x10^{-5}$ μCi/ml; and 4) total volume of effluent water. However, while the total amount of β-γ activity that is released is reported, we only list and identify that radioactivity by nuclide type in the report, if the activity exceeds $1x10^{-5}$ μCi/ml.

The original MITR-II Technical Specifications required the reporting for specific radionuclides if the total gross beta radioactivity exceeded 3E-6 μCi cm^{-3} at the point of release. This was presumably based on the old 10 CFR 20 (pre1994 – ICRP-2 methodology) (Appendix B, Table II, Column 2 for unidentified radionuclides whose half life exceed 2 hours).

The proposed Technical Specifications using the same logic should specify a requirement for the identification of radionuclides for reporting purposes that is

consistent with the new 10 CFR 20's (ICRP-30 methodology) MAC values (Appendix B, Table 3, Releases to Sewers). (Note: MAC is monthly average concentration.) The value for unidentified radionuclides with half-lives greater than 2 hours is 1 E-7 μCi cm^{-3}. In accordance with footnote 2 of Appendix B, a MAC of 1 E-5 is permitted provided that a mixture excludes certain radionuclides.

The majority of radionuclides present in a standard mix, for the MITR, represents MACS which are several orders of magnitude greater than 1 E-5. Therefore, a value of 1 E-5 μCi cm^{-3} is chosen as representative and conservative for the standard MITR mix. Again, this is only for listing nuclides by type in the report. All β-γ activity is reported.

77. Chapter 13, General: – Dutto and Evo used PARET to investigate step reactivity insertions. Both PARET and RELAP-5 were available for this purpose, and it was known that the two codes did not always give the same result. Therefore, rather than use PARET as a "black box," Dutto and Evo were careful to examine the assumptions that went into the code's formulation. Among the issues examined were time step size and adequacy of the code to treat two-phase conditions. Section II of their report summarizes these findings relative to the limitations of PARET and it is from that section (page 83) that the statement quoted in question 77 is taken. That statement refers to the correlations used in PARET to model two phase conditions. It does not refer to single phase conditions. If one reads p. 82, one realizes that Dutto and Evo did not state that the PARET equations were incorrect or even inadequate. Rather they stated that there might be issues arising from extending correlations that were obtained at steady-state to transients. They are recommending further research – a very reasonable position if one plans to use PARET for two-phase analyses. They also note (bottom of p. 82) that local voiding, if present, would lessen the predicted consequences of a step reactivity transient.

We do not feel that the quoted statement invalidates the use of PARET. Moreover, we note that the PARET code has been benchmarked using SPERT test data [ref. 13-12] and that a comparison of RELAP-5 and PARET by Woodruff, et al [Ref. 1] showed PARET to give better results in a benchmark comparison against the SPERT-IV data.

78. Chapter 13, General: – All transient calculations in Chapter 13 (loss of primary flow, ramp reactivity insertion, step reactivity insertion) have been redone using primary coolant flow of 1800 gpm, which is the LSSS. Please see response to questions No. 90 (ramp reactivity insertion); No. 92 (loss of primary coolant flow); and No. 80 (step reactivity insertion).

79. Chapter 13, General: – For purposes of clarification, the 71% figure cited in the question is obtained as (1800/2000)(.921)(.864). The first factor is the ratio of the LSSS flow to the normal primary flow. The latter two factors are as defined in

Section 4.6.3.2 of the SAR and were used in both the thermal-hydraulic limits calculations and the transient calculations. The transient calculations did use 2000 gpm as the initial contact flow. These have now been redone using 1800 gpm as noted in the response to question No. 78.

80. Chapter 13, General: – The Dutto and Evo report does not, as noted in the question, consider the corrosion film layer. Additional calculations have now been performed that do include the 2 mil corrosion layer. These results are summarized in Table One (next page) for both forced and natural convection. The approach taken to do these calculations differs from that in the SAR. In the SAR, we iterated on the magnitude of the step reactivity insertion in order to identify the addition that would cause the peak temperature to equal that of the softening point for aluminum. Here, we used the value previously determined as allowable in the SAR and showed that the softening temperature was not attained for these models.

Three models were evaluated. The first model assumed no fins (same as Evo and Dutto), minimum flow disparity, and 5 W initial power. For forced convection, the peak fuel temperature is calculated to be 432.07 °C. The second model included fin effects. It is not possible to model the fins with a one-dimensional code such as PARET. Thus, a model is created that preserves the salient features while modifying others. In this case, a fuel plate model was created that preserved both the surface heat transfer area between clad and coolant and the total masses of the clad and fuel. The dimensions of the fuel plate were not preserved (the width was increased but the thickness was decreased) and this in turn made the gap width (plate separation) less. In order to model the heat transfer correctly, thermal conductivities of the fuel and cladding were therefore reduced. The heat transfer coefficient of the modified fin model is about 87% of the no-fins model. Therefore the heat transfer and conduction conditions are similar in both cases except for the heat transfer surface area. The calculated peak temperature (338.91 °C) of the modified fin model was less severe than that of the first case (no fins).

The third case modeled the 2 mil corrosion layer. This was done by adding the corrosion layer to the model used in the first case (no fins). This choice was made because the first case had now been shown (432 °C vs 338 °C) to be the more conservative. The result was a calculated peak fuel temperature of 440.76 °C, still below the softening point of aluminum. Thus, we conclude that the step reactivity limits given in the SAR are acceptable even with allowance for a 2 mil crud layer. The same conclusion is evident for natural convection (please refer to the table).

81. Chapter 13, General: – Allowance was made for the corrosion layer in all analyses except that of step reactivity. The step reactivity case has now been done as well. Please see reply to question 80.

Table One

Summary of Step Reactivity Analysis for Question 80

	Initial Conditions	Step Reactivity Insertion	Max. Power	Peak Fuel Temperature	Over-power trip
Forced Convection					
1	No fins, min. flow disparity, initial power 5 W	$2.3	409.58 MW (0.147 s)	432.07 °C (1.50 s*)	7.4 MW (0.139 s)
2	Modified fin model, min. flow disparity, initial power 5 W	$2.3	393.86 MW (0.146 s)	338.91 °C (1.50 s*)	7.4 MW (0.139 s)
3	No fins, min. flow disparity, initial power 5 W, 2mils crud	$2.3	497.60 MW (0.148 s)	440.76 °C (1.50 s*)	7.4 MW (0.139 s)
*Simulation ends at t=1.50 s. Peak fuel temperature approaches equilibrium.					
Natural Convection					
4	No fins, min. flow disparity, initial power 5 W	$1.5	64.97 MW (0.352 s)	254.05 °C (0.390 s)	100 kW (0.113 s)
5	Modified fin model, min. flow disparity, initial power 5 W	$1.5	58.96 MW (0.349 s)	177.20 °C (0.356 s)	100 kW (0.113 s)
6	No fins, min. flow disparity, initial power 5 W, 2mils crud	$1.5	66.13 MW (0.353 s)	275.18 °C (0.390 s)	100 kW (0.105 s)

82. Section 13.1.1:

a) The range of the instrument is 1-100 μA with a typical steady-state 5 MW reading being about 70 μA. An oscillation of ± 1 μA would stand out as abnormal. Background noise is on the order of ± 0.5 μA.

b) No. There never has been any boiling of the primary coolant. However, one of the in-core experiments was a loop that simulated BWR conditions. The coolant contained in that loop was intentionally boiled and the effect was observable on the linear flux channel as oscillations of about ± 1 μA every few seconds.

c) The proper functioning of the detector is assumed (as it is for all of our nuclear instruments) through the periodic performance of functional checks and calibrations as specified by written procedures. We do not perform any special procedure for frequency response. (Note: Our use of the word, "high-frequency" is probably a misnomer and will be deleted. The oscillations observed in (b) above were every few seconds.)

d) Boiling of the type observed in the loop experiment described above can be sustained indefinitely without causing fuel failure. However, MITR Technical Specifications prohibit boiling of the primary coolant, and hence the operator should eliminate the condition as soon as possible.

e) If the cause of any observed noise is boiling, then a lowering of power will eliminate it. If some other factor is causing the noise, then it would likely continue despite a power reduction. So, this approach allows the operator to distinguish possible initiating events.

83. Section 13.1.2(a): There aren't any others that are credible. Also, we do not believe that even failure of an in-core experiment would create a step reactivity change. Such failures would occur as ramps which would be less severe.

84. Section 13.1.2 (b)(ii):

a) Variable speed is achieved by physically replacing one of the fixed-speed motors that drive the reactor control devices with a stepper motor. Overspeed protection is provided both through software and by a mechanical overspeed trip. Both are tested prior to the initiation of any digital control experiments.

b) No. The control experiments do not approach this value.

85. Section 13.1.6:

a) Please refer to Appendix A of Chapter 13 of the SAR. This appendix contains a synopsis of the safety analysis performed for one of the in-core loop experiments, the boiling coolant chemistry loop. In particular, please note Section A.4 which is a safety analysis. The scope of this analysis includes: temperature effects, hydrogen leak, loss of pumping power, in-thimble leakage, lead bath can leak, electric system failure, effect of boiling on reactor operations, and impact of the experiment on emergency core cooling. Analyses of this type are prepared for every in-core experiment, and are reviewed by the MIT Committee on Reactor Safeguards.

b) Yes, it is outside the scope of this section because the technical specification for step reactivity would not be exceeded.

86. Section 13.2.(a): Please see response to question 1.

87. Section 13.2.1.3: There appears to be a misunderstanding relative to this question. The over-pressure relief system does NOT initiate automatically. It must be placed on-line manually. This can be done without requiring entry to the reactor containment building.

a) None – The Maximum Hypothetical Accident (MHA) would not generate pressures that would require use of the pressure relief system.

b) Scenarios that could lead to use of the pressure relief system include two steps. First, the containment building is isolated for some reason. This could be the result of a radiological problem, a mechanical failure of the ventilation dampers, or an extended loss of off-site electricity. Second, the building heats up as the result of solar energy incident on the building. This process would take several hours.

88. Section 13.2.1.5: The thyroid dose indicated in this section is the 2-hour Committed Dose Equivalent (CDE).

89. Section 13.2.2.1: There are several parts to the question.

a) Fin Consideration: See reply to question 80.

b) Friction Losses: Analyses done for the case of no fins are conservative as shown in the reply to question 80. We have not analyzed, and see no need to analyze, the effect of corrosion layer buildup on hydraulic behavior given that the limiting case (no fins) is acceptable.

c) Two-Phase: Same reply as above for friction losses.

d) Natural Circulation: Please see response to question 80.

90. Section 13.2.2.2:

a) The ramp reactivity insertion limit, 6.5×10^{-4} ΔK/K/s that is given in the SAR was derived by using a kinetics calculation that investigated the effect of the power level and period scram set points. For a fixed initial power and period, the worst case was found to occur if the reactor scrammed simultaneously at 7.8 MW and on a 7-second period. The peak power, was found to be lower if the scram occurred only on power (7.4 MW) or only on period (7 seconds). For example, if the initial conditions were 6 MW and steady-state (infinite period), then a scram at 7.4 MW while on a period longer than 7 seconds would yield a lower-peak power. Similarly, a scram at 7 seconds while at a power less than 7.4 MW would yield a lower peak power. Our analysis did not include thermal hydraulic feedback. That is, no credit was taken for heat up of the coolant during the transient even though this would insert negative reactivity. The duration of the transient is so brief (a few seconds) compared to the transit time for the coolant (~30 s for hotter coolant to exit the core, move through the primary piping, and re-enter the core) that the effect of coolant heating would be minor. Once the peak power was determined, the calculation was completed by verifying that the safety limit (SL) was not exceeded for the assumed flow, temperature, and level conditions.

b) The analysis reported in the SAR showed an insertion rate of 6.5×10^{-4} ΔK/K/s to be acceptable. However, we took 5×10^{-4} ΔK/K/s as a limit. Additional calculations have now been performed for a ramp reactivity insertion of 5×10^{-4} ΔK/K/s for reactor power at 6 MW and 7.4 MW. The peak reactor powers that correspond to these initial conditions are 8.6 MW and 8.1 MW, respectively. These calculations were done assuming:

i) Reactor scrams when the reactor power or period reaches LSSS.

ii) There is an instrument delay time of 1 second following the initiation of reactor scram.

iii) The control blades start to insert at a rate of $1/second after the instrument delay. This reactivity insertion rate is conservatively derived from the 1-s blade drop time TS limit, and the minimum shutdown margin 1% ΔK/K ($1.27). As a comparison, the analyses described in section 13.2.2.2 of the SAR conservatively assumed that the reactivity insertion occurred after 2 seconds of instrument delay and control blade drop time. Therefore, no credit was taken for the negative reactivity insertion in deriving the ramp limit of 6.5×10^{-4} ΔK/K/s.

The results of these calculations are shown below:

	Initial Condition	Reactor Scrams	Peak Power
1	6 MW	High power scram at 7.4 MW (t=2.3 s, period = 8.5 s)	8.6 MW (t=3.6 s)
2	7.4 MW	High power scram (t=0.0 s)	8.1 MW (t=1.3 s)
3	5.3 MW (Worse case)	Scrams on high power (7.4 MW and short period (7-second at t=3.3 s)	8.9 MW (t=4.6 s)

The "worst case" initial reactor power for this transient is about 5.3 MW. At this initial power, both high power and short period scrams occur at the same time. The calculated peak power is 8.9 MW. We note that this worst case result is below the SL curve (Fig. 2.1.1 of TS) for the LSSS primary coolant flow and outlet temperature of 1800 gpm and 60 °C respectively. (Note: P/W_p is 49.4. The SL for 60 °C and 10 feet of coolant is about 50.8.) For initial power levels below 5.3 MW, the period scram halts the transient at a final power of less than 8.9 MW. For initial power levels above 5.3 MW, a power level scram at an LSSS halts the transient at a final power of less than 8.9 MW.

c) We are not aware of any credible accident that could generate a ramp reactivity insertion of 5×10^{-4} $\Delta K/K/s$.

91. Section 13.2.3.2:

a) The level of the coolant in the core tank would, as mentioned in Section 13.2.3.2, drop to three or four feet above the top of the core under worst-cases conditions (reflector dumped). If we assume an initial coolant height of 3 feet and limit the decay heat removal mechanism to evaporation only, we calculate that it would take 6 and 14 hours respectively for the coolant level to decrease to 2 feet and 1 foot above the core. The same calculation yields 23 hours for the top of the core to be uncovered. So there is ample time for a response.

b) The natural convection flow may be reduced because of the reduced pressure head above the core. However, this would not affect the above calculations because the only heat transfer mechanism considered for the analyses was evaporation.

c) The principal consequence is a loss of shielding which would affect the space above the reactor which is normally not occupied. This region is equipped with radiation monitors that alarm both locally and in the control room. It should be noted that the MITR is equipped with an emergency cooling system and this could be used to increase the water level.

92. Section 13.2.4:

a) Yes, there are check valves in the discharge lines of the primary coolant pumps that prevent backflow through an idle pump.

b) The loss of primary coolant flow was re-done for initial conditions of 7.4 MW, 1800 gpm, and 60 °C. The results are documented in the file memo, "Loss of Primary Coolant Flow Analysis," dated April 29, 2003. A copy is attached to this set of responses – See Appendix D. The basic finding of the revised calculation is that, as would be expected, the peak temperature is higher for the more stringent initial conditions. However, the difference is small and the peak temperature is still well below the 450 °C softening temperature of the clad.

c) An initial temperature of 60 °C was used in the revised calculations. Please see the above response.

d) The safety limit on reactor power for natural circulation operation is 250 kW. There is no safety limit for coolant temperature as explained in reply to question 33(c). Although the reactor decay power would exceed 250 kW during the first minute following a reactor scram, assuming a prior steady-state reactor power at 7.4 MW, the LOF calculation shows that the fuel temperatures are well below the softening point. This is because the primary coolant has adequate heat capacity so that nucleate boiling, which is calculated to occur only at the upper section of the hot channel, would not lead to onset of flow instability (Note: The natural circulation limits are not directly relevant to a LOF, because the analysis is for a critical reactor, which would not be the case following an LOF.)

93. Section 13.2.5.1: Please see reply to question 1

94. Section 13.2.6: "Experiment Malfunction" – Experiments that are conducted on the MITR are designed so as to preclude malfunctions that could either affect the reactor or the experimental facility. To that end, the general criteria listed in TS 6.1 as well as experiment-specific criteria that are developed in individual safety analyses are observed. Section 13.2.6 of the SAR divides experimental facilities into categories according to their physical location and then summarizes issues for which an accident potential exists. That discussion is expanded here:

a) Ex-Reflector Facilities – Experiments in those facilities are sufficiently distant from the core that they cannot cause a reactor accident.

b) In-Reflector Facilities – These are located in the re-entrant thimbles that penetrate the D_2O reflector tank and terminate near the light-water core tank. The concern here is that the thimbles might be damaged by

overpressure (explosion/radiolytic decomposition), corrosion, or internal heating. Reactivity effects from samples in these facilities are too small (a few millibeta) to be a credible cause of a reactor accident. Damage to the reactor from experiments in these facilities would not occur. But, loss of the heavy water, which contains tritium, is a possibility. Protection is provided by observing the general experiment requirements that are stipulated in TS 6.1.

c) In-Core Facilities – These are located in the core itself. The concerns here include:

- Ramp reactivity insertions should a facility flood or an in-loop sample move.

- Localized boiling of the reactor's primary coolant should excessive heat be produced in the facility and/or the facility lose cooling.

- Overpressure as the result of hydrogen buildup or uncontrolled production of steam.

- Release of chemical additives to the reactor's primary coolant should there be a slow leak.

The first of the above is addressed by observing the reactivity limits specified in TS 6.1. The others are precluded by the facility design which typically consists of an inner thimble that contains the experiment and which fits into an outer thimble. The outer surface of the latter is in contact with the reactor's primary coolant. The space between the two thimbles may be voided so as to minimize the coupling of the experiment to the core in all respects except neutronic. Or that space may be filled with a gas whose pressure is regulated so as to allow control of the heat transfer. Also, the inner thimble is provided with pressure reliefs and a safety system that monitors flow and temperature among other parameters within the inner thimble and which can shut down the power supply (electronic heaters) to the experiment.

95. Section 13.2.9.1: A new paragraph will be added at the end of Section 13.2.9.1. It will state:

> "Similar calculations have been performed with one blade full out. These calculations show a local increase in power of about 17%. Application of this factor to the hot channel [(44.8 kW/1.04)(1.17)] would make the power 50.4 kW which is again well below the safety limit of 67.9 kW."

96. Section 13.2.9.3: The reason for using 55 °C instead of 60 °C (LSSS for the primary coolant) is that the D_2O temperature is not linked to that of the primary. They are two separate systems, each with its own heat sink. Nevertheless, we

reviewed the calculation and, in so doing, found an error. For 55 °C, the evaporation loss is 102 liters, not 120 liters. We also redid the analysis for 60 °C. The evaporation loss is then 122 liters. Section 13.2.9.3 2 will be updated to use the 60 °C/122 liter figures.

97. Section 13.3: The 381 mrem at 21m is an error. The correct figure is, as stated in Section 13.2.1.5, 247 mrem at 21 m. Section 13.3 and Table 13.4 will be corrected to reflect this.

98. Chapter 15: Table 15-1 will be updated to include data from the most recent five fiscal years. The financial outlook for the MIT Research Reactor has improved recently because of the receipt of a funded INIE (Innovations in Nuclear Infrastructure and Education) grant for the U.S. Department of Energy. One other item of interest is that MIT redid the decommissioning cost estimate with a different contractor (Duke Engineering). The estimate was $20-25M, consistent with the previous estimate.

99. Section 16.3.1.4(a):

a) Radiation levels on the fuel preclude the performance of a surveillance that directly verifies the 2 mil thickness at the proposed fission density limit. However, other methods of surveillance are available and in use that will ensure safety. Some background information on this question is appropriate. MIT originally requested an increase in the fission density limit to 2.3×10^{21} ff/cc in 1990. On 01/19/91, the NRC requested additional information to support that request. MIT lacked the means to assemble the information that was required for a thorough response and therefore withdrew the request. In 1996/1997, a visiting French scholar (C. deWalsche from CEN Saclay, CEA) undertook a complete review of the available literature and prepared a 56 page report, "Prediction of the Oxidation of the Fuel Clad and Consequences for the MIT Research Reactor, June 1997." The study was done under supervision by senior NRL Staff as well as MIT NED faculty. De Walsche concluded that the Greiss correlation that had been used by MIT as the basis of its 1990 submittal was inappropriate and that the Kritz correlation gave more reliable results. He did 2D modeling of the fuel clad at an assumed MITR power of 10 MW and showed that the assumption of a 2 mil oxide layer at the 2.3×10^{23} ff/cc fission density limit was conservative. He also answered the NRC's 01/19/91 request for additional information. Those answers are provided here in Table Two (pp.35-36) together with the original questions. Given the above, we therefore conclude that:

i) The assumed 2 mil oxide layer is conservative for an MITR power level of 6 MW and a fission density limit of 2.3×10^{21} ff/cc.

ii) Sufficient uncertainty exists in the application of the Kritz correlation (as well as the other correlations) so as to warrant some means of surveillance to ensure safety. The uncertainty is the result of the empirical nature of these correlations and the fact that each application is different in terms of heat flux and water chemistry than the empirical data.

The surveillance that would be appropriate is the MITR's existing one whereby an air purge that is drawn from the air space below the reactor top lid and above the top of the primary coolant is monitored for abnormal radioactivity. If any is detected, then the fuel is "sipped." That is, the primary coolant is left undisturbed for 24-48 hours so that activity can build up. Water is drawn through each element and counted for iodine. This technique has proven very sensitive for the early detection of a clad problem. The monitor is already required by TS 3.7.1.3.

b) Not applicable. Please see reply to question 1.

100. <u>Section 16.3.1.7:</u>

a) This is not an issue because only helium is used as a blanket gas in the reflector region. At the time of submission of the relicensing materials in 1999, a study was in progress to determine if we could replace the helium cover gas with CO_2. The study concluded that this would not be advisable because of: 1) the moisture content of the CO_2 which would lead to carbonic acid production as noted in the question; and 2) the purity of CO_2 in bulk quantity. The sixth sentence of Section 16.3.1.7 will be changed to delete the words "or non-reactive (CO_2) gas."

b) No inspections of the outer surface of the interior tank surface are planned as regular surveillance.

Table Two

Response to NRC Request for Additional Information,
Dated 01/19/91*

1. ***Compare directly the predicted oxide thickness for extended burn-up with the oxide thickness assumed in the FSAR for the presently approved burn-up. Discuss whether the new predictions lead to fuel temperatures above limits previously analyzed and approved for normal operation.***

Response:

According to the present study, a maximum thickness of 1.9 mils for the oxide layer thickness should not be exceeded in the MITR-III. According to Griess, no risk of spallation or clad deterioration is to fear below this limit. The calculations used in the Kritz correlation with the correcting factor proposed by Griess to take into account the high pH of MITR and is valid for a low flow rate hypothesis (2500 gpm) and a high core inlet temperature (60 °C) hypothesis. The calculation took into account the increase of the wall temperature due to the oxide formation. The maximum temperature increase obtained was inferior to 4 °C and the wall temperature never exceeded 94 °C in the hot channel. The 2 mil limit to prevent risks of spallation is probably very conservative though. Indeed, according to recent studies, spallation is linked to thermal stress and should not occur for a temperature drop of less than 113 °C in the oxide layer, a value which will never by reached in MITR as the heat flux is too low (1MW/m^2 at the hot spot of for a 10 MW power). A maximum clad temperature increase of 15 °C must be expected in the presence of a 2 mil oxide thickness, according to two-dimensional complementary results, so that the clad temperature will remain very far from the integrity temperature limit ([redacted].

2. ***Oxide thickness also affects responses to rapid insertions of reactivity, and perhaps, other MIT accident scenarios. Please review and re-analyze all potential accidents and discuss whether FSAR conclusions would remain valid with the projected increases in oxide thickness.***

Response:

It is usually assumed for transient analyses that the clad is a homogenized non-finned plate. If the same hypothesis is assumed in the presence of a 2 mil oxide layer, the oxide influence should be relatively small and would even lead to lower peak temperatures in the case of a reactivity insertion scenario. Yet, two-dimensional modeling revealed that the influence of the oxide layer should lead to higher temperatures. For a 2$ reactivity insertion scenario, the peak temperature was estimated to be 290 °C, while it was around 250 °C for a clad with no oxide. The obtained temperatures remain far away from the clad structural integrity limit of [redacted], though.

Table Two (Continued)

3. ***The increased oxide thickness will decrease the hydraulic diameter of the grooves. This will result in increased pressure losses due to friction and to decreased coolant velocities in the grooves. Please provide analyses of the impact of these changes on hot channel factors, and assess to what extent the decreased coolant velocities affect the oxide buildup or other crud deposition in the grooves. Unless justification can be provided that grooves do not become clogged, please provide analyses of fuel temperature conditions both in steady state and potential accident scenarios with the grooves filed with oxide.***

Response:

In the worst case (no dissolution of the reacting aluminum in the coolant), a 2 mil boehmite layer would represent a reduction of the hydraulic diameter by 2.2%. This is a very conservative value, though, as a part of the reacting aluminum should be dissolved in the coolant. On the same assumption, the space between the fins would be reduced from 10 mils to 8 mils (>200 μm). As the boundary layer thickness is less than 5 μm for MITR-III, it is expected that this reduction would have a very limited influence on the coolant velocity in the groove.

4. ***The thermal conductivity assumed for the oxide on the fuel plates appears to be inconsistent. The response of the request for information dated 11/28/89 states a thermal conductivity of 2.0 Btu/hr-°F-ft. The conductivity used will influence fuel plate temperatures, transient response to accidents, and additional oxide growth since the oxide-aluminum interface temperature controls oxide growth. Please justify the use of the 2.0 Btu/hr-°F-ft value in your analyses, or re-analyze reactor behavior with the Griess value of 1.3 Btu/hr-°F-ft.***

Response

All the present study assumed the Griess value of 1.3 Btu/hr-°F-ft for the conductivity of the oxide.

*The above was taken from report written by C. deWalsche, 1997.

The following responses concern the Technical Specifications

101. TS 1.3.32.4: We believe that the question refers to TS 1.3.32.5, not 1.3.32.4. TS 1.3.32.5 will be changed to be consistent with ANSI/ANS-15.1-1990 Section 6.7.2(1)c(iv) and it will read "an uncontrolled or unanticipated change in reactivity greater than 0.78% ΔK/K ($1.00)."

102. TS 2.1.1: No change is necessary. Please see response to question 29. The value of R is 0.86 and Figure 4.25 (same as Fig. 2.1-1 in TS) is correct.

103. TS 2.2: No change is necessary. Please see response to questions 31-33. In particular, the equations that form the basis of TS 2.2 are valid.

104. TS 3.1.2: We do not see any inconsistencies between TS 3.1.2 and TS 3.1.4.5. The MITR has six shim blades. TS 3.1.4.5 requires that there be at least 5 operable blades and that any inoperable ones be at or above the shim bank height. TS 3.1.2 specifies the shutdown margin (SDM) and requires, among other things, the most reactive operable blade is fully withdrawn. Therefore, if all six blades are operable (normal case), the SDM is calculated assuming the most reactive of the six blades is stuck out. If only five blades are operable, then the SDM is calculated assuming that both the inoperable blade and the most reactive of the five operable ones are stuck out. The existing language already covers all possibilities and we feel that no change is needed.

105. T.S. 3.1.3: Please see response to question 89. No change is needed to the specification. The relevant section of the SAR will be updated to include the analysis given in question 89 and it may then be appropriate to update the basis.

106. TS 3.1.4.4(a): The value of R is correct as are the resulting equations. No change is needed. See response to question 102.

107. TS 3.1.4.4(b): Equations (4-40) and TS Equation (2.2-1) are correct. No change is necessary.

108. TS 3.1.4.4(c): The evaluations are performed by the reactor engineer, checked for proper documentation by the Q/A Supervisor, and reviewed by the Director of Reactor Operations. See Section 12.1.2.1 for descriptions of the duties of these individuals. Individuals who are designated as "Reactor Engineer" hold (or have held) an SRO License at the MITR and possess (or are pursuing) an advanced degree in nuclear engineering (or a closely related field). In addition, the designated individual(s) must have demonstrated capability for performing this type of calculation to the MITRSC. The last line of TS 3.1.4.4(c) will be changed to read, "and approved by the reactor engineer and the Q/A Supervisor." Also, a new paragraph (f) will be added to Section 12.1.4 of the SAR to state:

f) Reactor Engineer – Individuals designated as a reactor engineer shall have a recognized baccalaureate degree in an engineering or scientific field be designated by the MIT Reactor Safeguards Committee as competent to perform refueling calculations for the MITR.

109. TS 3.1.4, basis (4(a): The value of the safety limit in TS 3.1.4 basis 4(a) should be 9.1 not 9.0. Please see response to question 59.

110. TS 3.1.6.3: Not applicable. Please see response to question 1.

111. TS 3.1.6, basis 3:

a) See reply to question 80.

b) 2.08 J/m K

112. TS 3.2.2.4: The 0.5% ΔK/K (636 mbeta) figure is appropriate because: (1) it is low enough so that the safety system is capable of protecting the fuel against such an insertion; and (2) it is high enough so that a practical automatic control system can be designed. The justification of this figure in terms of moveable, non-secured, and secured experiment criterion is inappropriate. The last three sentences of the basis will be condensed to read, "the value chosen for the reactivity worth limitation is 0.5% ΔK/K which is well below the step reactivity insertion limit."

113. TS 3.2.3.3: The symbols in Table 3.2.3-1 will be changed so that only "less than" or "greater than" symbols are used.

114. TS 3.2.4.1(a): Please refer to Section 2.2.2.1 (p. 7-7) of the SAR. The startup interlocks are part of the withdraw permit circuit which consists of more than 30 individual relays. The first four of these relays are referred to as the "startup interlocks." The others are individual scrams. The four that comprise the "startup interlocks" are not safety related and are intentionally bypassed upon initiation of startup. Their purpose is solely to ensure "that operating conditions are stable prior to startup."

115. TS 3.2.7: The indication may be either numeric or analog meter or both as noted in Table 3.2.7-1 of the TS. Also, specified is the location where the indication shall be visible. The most essential items are displayed on the console. Indication is verified prior to startup by the performance of the instrumentation and mechanical startup checklists.

116. TS 3.3.1.2: The mechanical startup checklist requires that the closing of these valves be observed either visually (normal method) or audibly (requires special equipment) upon the starting of the primary coolant pumps. Similarly, the

mechanical shutdown checklist requires that the opening of these valves be verified once the pumps are secured.

117. TS 3.1.3 and SAR Section 13.2.2.1: The question is correct in that initial conditions of high and low power were assumed respectively for forced and natural convection. We have now done analyses (see reply to question 80) for a forced convection low power (5 W) condition.

118. TS 3.3.2: The Standing Order remains in effect and has recently been re-affirmed and discussed with all licensed personnel. Additional measurements of H_2 production as a function of power level have been performed since the initial submission of the MITR relicensing materials. These confirm the need for the Standing Order, although they also show it to be overly restrictive. Specifically, the most recent results indicate that an H_2 concentration of 3.5% will be attained in 13 minutes after isolation of the air space above the core at 4.9 MW. For 6 MW, the time is estimated to be 12 minutes. Technical Specification No.3.3.2.2 will therefore be changed to read, "In the event of isolation of the air space above the core for more than five minutes, reactor power shall be reduced to <100 kW." The basis of the specification will also be revised accordingly.

119. TS 3.3.6: Conductivity is easily measured and there is a continuous display (with alarms) of conductivity in the reactor control room. An abnormal conductivity may indicate a problem such as out-of-specification pH or high chloride ion concentration or it may indicate something abnormal but nevertheless benign such as a colloidal suspension of some of the metal oxide. (Note: The latter is mentioned solely for purposes of illustration. The MITR has never had such an occurrence.) Hence, there is no time limit on the out-of-specification conductivity provided that the requirements of TS 3.3.6.3, which address pH and chlorides, are met.

120. TS 3.5.1 and 3.5.2: The purpose of both specifications is to minimize the generation and build-up of argon-41. There are no corresponding specifications in the current MITR technical specifications. However, there are abnormal operating procedures that establish this practice. The reason for the selection of 250 kW is that experience has shown argon generation to be minimal below this level. Also, this is the power level at which MITR startup procedures specify that the cooling tower be placed on line. Hence, its selection is also a matter of convenience. The new paragraph will be added to the basis of TS 3.5 that states, "Argon-41 production is minimal below 250 kW. Hence, reactor power shall not be raised above 250 kW unless ventilation has been established and reactor power shall be reduced to less than 250 kW within five minutes if ventilation is lost."

121. TS 3.7.1.3: The TS will be modified to add the following: "if the preceding is done, then the portable instrument shall be read and/or the survey or analysis performed at least daily. If there are elevated readings on the plenum effluent monitors, then the frequency shall be at least every eight hours." The core purge

effluent goes through the exhaust plenum and hence any abnormal activity would be detected on the redundant plenum monitors that indicate continuously in the control room and that are alarmed.

122. TS 3.7.4: TS 3.7.4 covers only the receipt of byproduct material on the reactor license. It does not cover special nuclear or source material. The MITR license currently authorizes possession of two 1 curie Pu-Be sources, one 150 Curie Sb-Be source, the fuel needed to operate the MITR, and the associated fission products. The proposed possession limits are the same.

123. TS 4/TS 1.3.11: The waiver contained in TS 1.3.11 is intended to apply to the specifications that cover experiments. In particular, TS 6.4, TS 6.5, and TS 6.6 (if fuel is removed). The intent was to eliminate the need to perform test/calibrations of equipment that was not in use and had no need to be in use. The waiver does not apply to any of the TS 4 requirements. The third to last line of TS 1.3.11 will be modified to read, "Surveillance tests required for experiments (Section 6) may be waived when...."

124. TS 4.1.5(b): The fuel depletion methodology for the MITR is described in an 819 page document, "MITR-II Fuel Management, Core Depletion, and Analysis: Codes Developed for the Diffusion Theory Program CITATION" [Ref. 2]. The error analysis is covered in pp. 258-285. The principal sources of error in the determination of the fission density are the initial U-235 content of the element, the core flux distribution, and the net energy production by the reactor. These errors were determined to ten, ten, and five percent respectively (MITR-II SAR). These errors are combined statistically because fuel is rotated, shuffled, and inserted and hence no single element is always worst case. The result is a net error of 15%. Hence, the criterion for removing a fuel element is 85% of the fission density limit. Higher burnups are allowed if the uncertainties in the initial fuel loading, flux distribution, and/or energy output can be shown to be less. FYI, the error analysis for the MITR fuel depletion was independently reviewed by a non-MIT professional engineer.

125. TS 4.2.1: The preferred approach is measurement. However, it is desired to have an alternative in the event that a measurement is not immediately feasible. To that end, we have assembled correlations for the following:

a) Effect of Core Radial Power Distribution on Blade Reactivity: The core radial power distribution, as determined from 3D full-core computer models, has been correlated with integral shim bank worth. This allows the prediction of the effect of a major refueling on bank worth. [Ref. 3]

b) Effect of Core Burnup on Blade Reactivity: The effect of overall core burnup, as measured by energy prediction (MWDs), has been correlated with integral shim bank worth. This allows the prediction of the affect of a blade changeout on bank worth. These data have not been published.

Both correlations are accurate to about 10%, which is similar to the accuracy of a measurement.

126. TS1.3.3.7, TS 3.2, and TS 4.2: The definitions are all intended to be the same. TS 1.3.3.7 is a definition. TS 3.2.2. is a requirement that is imposed on the reactor control and safety system. Both TS 1.3.3.7 and TS 3.2.2 use the same language. TS 4.2.3 provides the surveillance criteria that are to be used to demonstrate compliance with TS 3.2.2. In order to do that, it specifies that the measurement be made from the full out position. This is a conservative requirement.

127. TS 4.2.4: The basis for the 24 hour shutdown limit is ANSI/ANS 15.1-1990, Section 4.2(5)(a) which is on p. 6 of the standard. There were 82 shutdowns of less than 16 hours during calendar year 2002. The current MITR TS require that these checks be done monthly and each time before startup if shutdown more than 16 hours. The change to quarterly and 24 hours is to be made in accordance with the above ANSI/ANS standard.

128. TS 4.3.5: The reference made in this specification to sampling requirements in TS 3.3.2 and 3.3.3 is an error. Such sampling requirements exist in the current version of the MITR Technical Specifications to cover the situation where the space above the core is isolated and reactor operation at power is continued. As explained in the response to question 118, that situation will no longer exist. Also, there is no rationale for the annual sampling requirement. The D_2 is in a helium atmosphere and we have been unable to identify an instrument that will reliably measure D_2 in anything except air. We have a recombiner together with a technical specification that requires its operability and that recombiner insures the absence of D_2 in the gas space above the reflector. For H_2, we now have extensive base-line data and the production rate will not change. Accordingly, this specification will be deleted.

129. TS 5.3: Specifications for the MITR fuel are given by a DOE-INEEL document, "TRTR-3, Rev. A, May 5, 1979 as amended." This document is entitled, "Specification for Massachusetts Institute of Technology Fuel Elements." It is about 80 pages including appendices. This document specifies fin dimensions as well as many other parameters. The manufacturer's quality assurance (Q/A) people verify each plate (and later each element) against those specifications. MIT repeats certain of these Q/A checks upon receipt of the fuel. We are not in a position to verify fin dimensions because once the element is assembled, we only have access to one side of the two exterior plates. ([redacted] There is no need to include such a specification on fin dimensions in the TS because it already exists in the DOE document and is checked by the manufacturer's Q/A people. Also, there would be no advantage to including such a specification because MIT cannot verify it and NRC cannot inspect it. (Note: MIT does visit the manufacturer's plant periodically and we do verify fin measurements during those

visits. Also, we receive reports of any defective plates and review these reports for trends.)

130. TS 5.3: Please see response to questions 80 and 89. The statement at the end of the second paragraph of the basis is intended to provide a generic justification for using as thin a clad as possible. It was not intended nor should it be interpreted as the result of an analysis.

131. TS 6.1.1: The specification is in fact administered as described by the question. The reactivity worths for single experiments are treated as absolute values and the total is the sum of those values. An additional footnote will be added to the table that appears in TS 6.1.1. It will state, "Total worths are to be determined by summing the absolute value of the reactivity worth of each single experiment."

132. TS 6.1.4.(c): The limit is per capsule. TS 6.1.4 (c) will be modified to add an additional sentence: "The total number of vented capsules shall be limited so that the limits of 10 CFR 20 at any point of possible exposure are not exceeded."

133. TS 7.6.2(a): The phrase will be changed to read, "... no immediate safety significance to the reactor." Its purpose is to allow continued operation of the reactor in the event of occurrences that do not affect the reactor such as ones arising from an ex-core experiment.

134. TS 7.8.1(c): Paragraph (f) will be added to TS 7.8.3. It states: "f) Records of reviews of violations of limiting conditions for operation (LCOs)"

135. TS General: Six amendments have been issued to the MITR Technical Specifications since July 1999. These are:

Number	Date	Purpose
N/A	08/10/99	Corrected minor error in the basis of TS 3.10(3)(e) which covered the storage of spent fuel.
31	12/21/99	Added TS 6.6 (fission converter).
32	04/02/01	Updated TS 6.5 (human therapy using the fission converter).
33	05/30/02	Updated TS 7.1 (administrative re-organization of Reactor Radiation Protection Office).
34	03/25/03	Modified TS 6.6.2.6 and 6.6.3 (fission converter surveillances).
35	04/16/03	Added TS 6.7 (in-core fissile experiments) Modified TS 6.1.7 (experiment radiation release)

The above will be incorporated in the renewed technical specifications when we incorporate the response to the other 134 questions listed above.

References

1. Woodruff, W. L., N. A. Hanan, and J. E. Matos, "Comparison of the RELAP5/MOD3 and PAARET/ANL Codes with the Experimental Transient Data

from the SPERT-IV D-12/25 Series," Presented at the 1997 International Meeting on Reduced Enrichment for Research and Test Reactors, Oct. 1997.

2. Bernard, J.A., "MITR-II Fuel Management, Core Depletion and Analysis: Codes Developed for the Diffusion Theory Program CITATION," MS in Nuclear Engineering Thesis, June 1979.

3. Bernard, J.A., Kwok, K.S., Lanning, D.D., and L. Clark, Jr., "Effect of Radial Power Distribution on MITR-II Fuel Element and Control Blade Worth," Transactions of the American Nuclear Society, Vol. 49, Suppl. 2, Aug. 1985, pp 53-54.

Appendix A

Additional Information for Response to Question 41

RESPONSE TO QUESTION 41 OF THE SECOND PARTIAL REQUEST FOR ADDITIONAL INFORMATION

MASSACHUSETTS INSTITUTE OF TECHNOLOGY RESEARCH REACTOR DOCKET NO. 50-20

QUESTION 41:

Section 5.5.1.5, "Radiation Monitors," Page 5-47. Please discuss the amount of radioactive material that could enter the secondary system due to a potential heat exchanger failure before detection by the radiation monitors or sampling. What would be the path of the radioactive material to the environment and what would the maximum potential dose be to a member of the public?

RESPONSE:

Introduction:

Assuming a condition capable of inducing a leak of radioactive material into the secondary system, the radionuclides capable of producing the largest potential dose to a member of the public are H-3 and Na-24. There are two possible paths they could take to the environment.

1) H-3 and Na-24 in the primary coolant (H_2O) leak into the secondary system, and from there, leave via evaporation (H-3 only) or blowdown.

2) H-3 and Na-24 in the reflector heavy water (D_2O) leak into the secondary system, and from there, leave via evaporation (H-3 only) or blowdown.

For either of these cases, it is prudent to consider two main release scenarios.

1) The leak is so small that it is essentially "never" detected.

2) The leak is large enough that it will be detected.

Removal Constants for H-3 and Na-24:

The first step of the analysis is to calculate the removal constants for H-3 and Na-24.

For H-3:

$$k_H = \lambda_H + k_{BD} + k_{Evap}, \quad (1)$$

where

k_H = total H-3 removal constant, radioactive decay plus physical removal (min^{-1}),
λ_H = H-3 radioactive decay constant (min^{-1}),
k_{BD} = blowdown removal constant (min^{-1}), and
k_{Evap} = evaporation removal constant (min^{-1}).

Substitute the following numerical values into Equation 1 to calculate k_H:

λ_H = 1.07×10^{-7} min^{-1},
k_{BD} = 1.85×10^{-3} min^{-1} (= blowdown rate of 4.41×10^4 mL/min divided by total secondary system water volume of 2.38×10^7 mL), and
k_{Evap} = 4.23×10^{-3} min^{-1} (= evaporation rate of 1.01×10^5 mL/min divided by total secondary system water volume of 2.38×10^7 mL).

For Na-24:

$$k_{Na} = \lambda_{Na} + k_{BD}, \quad (2)$$

where

k_{Na} = total Na-24 removal constant, radioactive decay plus physical removal (min^{-1}),
λ_{Na} = Na-24 radioactive decay constant (min^{-1}), and
k_{BD} = blowdown removal constant (min^{-1}).

Note: there is no Na-24 loss via evaporation.

Substitute the following numerical values into Equation 2 to calculate k_{Na}:

λ_{Na} = 7.73×10^{-4} min^{-1}, and
k_{BD} = 1.85×10^{-3} min^{-1} (= blowdown rate of 4.41×10^4 mL/min divided by total secondary system water volume of 2.38×10^7 mL).

The calculated removal constants for H-3 and Na-24 are summarized in Table 1.

Table 1: Removal Constants for H-3 and Na-24

Nuclide	Total Removal Constant (min^{-1})
H-3	6.1×10^{-3}
Na-24	2.6×10^{-3}

H-3 and Na-24 Released if Leak is Undetected:

The first of the two main release scenarios is one in which the leak is so small that it is essentially "never" detected. For the discharge to remain completely unnoticed over a long period, it must be small enough to escape detection by even the most sensitive method for determining leaks, which for both H-3 and Na-24 is daily secondary system water sampling and analysis by Reactor Radiation Protection Technicians. The total amount of both H-3 and Na-24 released over the course of one year, due to an undetected leak in either the primary coolant system or the reflector heavy water system, is calculated as follows.

H-3 Released in One Year Due to an Undetected Leak:

Assuming steady-state conditions, the following equation represents the tritium concentration in the secondary water:

$$C_H = \left(\frac{1}{V}\right)\left(\frac{C'_H R}{k_H}\right), \tag{3}$$

where

C_H = steady state H-3 concentration in the secondary system water (μCi/mL),
V = total secondary system water volume (mL),
C'_H = initial H-3 concentration in either primary or the reflector (μCi/mL),
R = leak rate from either primary or the D_2O reflector into secondary (mL/min), and
k_H = total H-3 removal constant, radioactive decay plus physical removal (min^{-1}).

Solving Equation 3 for *R* yields:

$$R = \frac{C_H V k_H}{C'_H}. \tag{4}$$

Substitute the following numerical values into Equation 4 to calculate R:

C_H = 4.5×10^{-5} μCi/mL, the minimum concentration detectable by daily sampling,
V = 2.38×10^{7} mL,
k_H = 6.08×10^{-3} min^{-1}, and
C'_H = 3.2×10^{-2} μCi/mL for primary *or* 4.15×10^{3} μCi/mL for reflector (the current average values for a one-year period).

If the calculated steady-state leak rate (R) into the secondary system is either less than about 203 mL/min from primary, or less than about 1.6×10^{-3} mL/min from the D_2O reflector, the tritium concentration in secondary will remain less than 4.5×10^{-5} μCi/mL (used as C_H, above). At this concentration of H-3 in a daily secondary water sample, a Radiation Protection Technician using liquid scintillation counting could reasonably be expected to notice a change that requires further investigation.

Because a concentration of 4.5×10^{-5} μCi/mL tritium in the secondary water is the upper limit for undetected leaks from *either* primary *or* the reflector, the total activity released to the environment in one year for each case will be identical. Simply multiply the threshold concentration (4.5×10^{-5} μCi/mL) by the total amount of secondary water discharged in one year via either blowdown (4.41×10^{4} mL/min x 5.256×10^{5} min) or evaporation (1.01×10^{5} mL/min x 5.256×10^{5} min) to determine the total annual discharge.

Note that for the case of a release from primary, this is a very conservative estimate, because at a leak rate of 203 mL/min, the primary storage tank low-level alarm would actuate within about 18 hours (alarm occurs at a loss of 57 gallons). Thus the leak *would be* detected, and would not

continue unabated for an entire year.

The calculated values for total H-3 activity released in one year are summarized in Table 2.

In addition to the total activity released, the average tritium concentration of a release can be determined. No calculation is needed to determine the concentration of tritium in the blowdown water, because it is assumed to be at the threshold concentration (4.5×10^{-5} μCi/mL) for detection by a Radiation Protection Technician. However, the tritium in the evaporated water will be released to the atmosphere. To calculate the H-3 concentration in *air* resulting from evaporative discharge from the cooling towers, multiply the annual average water vapor content in Boston air, 6.8×10^{-6} mL H_2O / mL air (MITR-II Safety Analysis Report, Section 12.1.2.2), by the threshold secondary water concentration of 4.5×10^{-5} μCi/mL.

The calculated values for average tritium concentration of a release are summarized in Table 2.

Table 2: H-3 Activity Released in One Year Due to an Undetected Leak

Discharge Path	Annual H-3 Activity Released	Concentration of Release (Average)
Blowdown (via Primary to Secondary)	1.0×10^{6} μCi	4.5×10^{-5} μCi/mL H_2O
Evaporation (via Primary to Secondary)	2.4×10^{6} μCi	3.1×10^{-10} μCi/mL air
Blowdown (via Reflector to Secondary)	1.0×10^{6} μCi	4.5×10^{-5} μCi/mL H_2O
Evaporation (via Reflector to Secondary)	2.4×10^{6} μCi	3.1×10^{-10} μCi/mL air

The undetected primary blowdown release over the course of one year may actually represent just a *part* of the known discharge each year, rather than a hypothetical additional release that must be calculated separately. However, even if this undetected activity is added as an "extra" to the known yearly liquid effluent discharge amount of 265 mCi (FY 2002 Annual Report), the total release to sewers would still only be about 1.3 Ci (1.0×10^{6} μCi + 265 mCi) for the entire year. Also, because the threshold concentration for the H-3 is 4.5×10^{-5} μCi/mL, the Technical Specification requirement for a secondary water tritium concentration less than 1 μCi/L (=1.0×10^{-3} μCi/mL) will clearly be satisfied. Also, the H-3 concentration of the discharged water will be less than the Monthly Average Concentration (MAC) for tritium (1.0×10^{-2} μCi/mL per 10 CFR 20).

The undetected evaporative release via primary may also represent just a *part* of the known discharge each year, rather than a hypothetical additional release that must be calculated separately. However, even if this activity is added as an "extra" to the known yearly evaporative discharge amount of 0.641 Ci (FY 2002 Annual Report), the total evaporative release would still only be about 3.0 Ci (2.4×10^{6} μCi + 0.641 Ci) for the entire year. Even under these conditions, the H-3 concentration in air would not exceed the limit of 1.0×10^{-7} μCi/mL air (10 CFR 20).

Na-24 Released in One Year Due to an Undetected Leak:

Assuming steady-state conditions, the following equation represents the sodium concentration in the secondary water:

$$C_{Na} = \left(\frac{1}{V}\right)\left(\frac{C'_{Na} R}{k_{Na}}\right), \tag{5}$$

where

C_{Na} = steady state Na-24 concentration in the secondary system water (μCi/mL),
V = total secondary system water volume (mL),
C'_{Na} = initial Na-24 concentration in either primary or the reflector (μCi/mL),
R = leak rate from either primary or the D_2O reflector into secondary (mL/min), and
k_{Na} = total Na-24 removal constant, radioactive decay plus physical removal (min^{-1}).

Solving Equation 5 for *R* yields:

$$R = \frac{C_{Na} V k_{Na}}{C'_{Na}}, \tag{6}$$

Substitute the following numerical values into Equation 6 to calculate R:

C_{Na} = 1.0×10^{-6} μCi/mL, the minimum concentration detectable by daily sampling,
V = 2.38×10^{7} mL,
k_{Na} = 2.62×10^{-3} min^{-1}, and
C'_{Na} = 5.18×10^{-1} μCi/mL for primary or 1.49×10^{-3} μCi/mL for reflector (the current average values for a one-year period).

If the calculated steady-state leak rate (R) into the secondary system is either less than about 0.1 mL/min from primary, or less than about 42 mL/min from the D_2O reflector, the sodium concentration in secondary will remain less than 1.0×10^{-6} μCi/mL (used as C_{Na}, above). At this concentration of Na-24 in a daily secondary water sample, a Radiation Protection Technician using gamma spectroscopy could be expected to notice a change that requires further investigation.

The Na-24 threshold concentration of 1.0×10^{-6} μCi/mL represents the concentration of blowdown water for undetected leaks from *either* primary *or* the reflector (see Table 3). Therefore, the total activity released to the environment in one year for each case will be identical. Simply multiply the threshold concentration (1.0×10^{-6} μCi/mL) by the total amount of secondary water discharged in one year via blowdown (4.41×10^{4} mL/min x 5.256×10^{5} min) to determine the total annual discharge.

Note that for the case of a release from the reflector, this is a very conservative estimate, because at a leak rate of 42 mL/min, the reflector dump tank low-level alarm would actuate within 12 hours (alarm occurs at a loss of 8 gallons). Thus the leak *would be* detected, and would not

continue unabated for an entire year.

The calculated values for total activity released are summarized in Table 3.

Table 3: Na-24 Activity Released in One Year Due to an Undetected Leak

Discharge Path	Annual Na-24 Activity Released	Concentration of Release (Average)
Blowdown (via Primary to Secondary)	2.3×10^4 μCi	1.0×10^{-6} μCi/mL
Blowdown (via Reflector to Secondary)	2.3×10^4 μCi	1.0×10^{-6} μCi/mL

The Na-24 concentration in the secondary water will clearly remain below the MAC of 5.0×10^{-4} μCi/mL (10 CFR 20).

H-3 and Na-24 Released During a Detected Leak:

If the leak is large enough that it *will be* detected, the release will occur over a much shorter time span, and there is the potential for a much larger discharge. The total activity released to the environment can be limited by any one of the following three factors.

1) *Daily Secondary System Water Sampling*

   A 2 mL sample of secondary water is analyzed daily for tritium content using a liquid scintillation detector. As discussed, the concentration at which a Radiation Protection Technician will investigate further is 4.5×10^{-5} μCi/mL. Assuming that the leak rate is large enough to build in a concentration at least that high, then no more than 24 hours will elapse before the leak is noticed (thus the upper limit for t in Equations 7-10, 12, and 13 is 24 h = 1440 min).

2) *Secondary System Water Monitors*

   These two monitors consist of Na(Tl) crystals (coupled to photomultiplier tubes) that detect gamma-emitting radionuclides. Of primary interest is Na-24, which builds up in both the primary water and in the reflector D_2O when the reactor is operating. If the Na-24 concentration in the secondary water becomes high enough (2.7×10^{-4} μCi/mL), an alarm will be actuated in the Control Room. Thus, although tritium itself will not cause the secondary water monitors to respond, the Na-24 that will be released in conjunction with the H-3 will serve as an indirect indicator. The Na-24 exponential buildup with time is expressed as follows:

$$C_{Na} = \left(\frac{1}{V}\right)\left(\frac{C'_{Na} R}{k_{Na}}\right)\left[1 - e^{-k_{Na} t}\right] \leq C_A, \qquad (7)$$

where

C_{Na} = Na-24 concentration in the secondary system water (μCi/mL),
V = total secondary system water volume (mL),
C'_{Na} = initial Na-24 concentration in either primary or the reflector (μCi/mL),
R = leak rate from either primary or the D_2O reflector into secondary (mL/min),
k_{Na} = total Na-24 removal constant, radioactive decay plus physical removal (min^{-1}),
t = elapsed time (min); with a maximum value of 1440 min, and
C_A = Na-24 concentration at which an alarm will be actuated (2.7×10^{-4} μCi/mL).

Note that other gamma-emitting radionuclides will be present in the water, as well, causing a secondary monitor response sooner than if Na-24 were the only gamma-emitter present. However, no credit is taken for this in the calculations (a conservative assumption).

3) *Primary Coolant and/or Reflector D_20 Height*

It is prudent to assume that the Console Operator in the Control Room will not become aware of a leak until an alarm is actuated. A leak in the primary system will cause a low primary storage tank alarm after a 2" (57 gallon) drop in the tank water level. Similarly, a leak in the D_2O reflector will cause a low D_2O dump tank alarm after a 2" (8 gallon) drop in the tank heavy water level. Conservatively assuming that the operator will not take action until a drop equal to twice the alarm setpoint value (4"), the maximum discharge from primary to secondary will be approximately 114 gallons, and the maximum discharge from reflector to secondary will be 16 gallons. Thus the product of the leak rate and the period during which the leak occurs cannot exceed the limiting volume at which an operator would take corrective action to prevent further release:

$$Rt \leq V_T, \tag{8}$$

where

R = leak rate from primary or the D_2O reflector into secondary (mL/min),
t = elapsed time (min); with a maximum value of 1440 min, and
V_T = volume lost from either primary storage tank or D_2O reflector dump tank at which an operator will take corrective action (mL).

H-3 Released During a Detected Leak:

The total H-3 activity released (via either blowdown or evaporation) during a leak can be calculated by integrating an activity release rate. The lefthand side of Equation 9 contains an additional physical removal constant (k_R) that transforms the quantity being integrated from an atom release rate into an activity release rate.

The total H-3 activity released during a leak will be as follows, with the caveat that the upper limit of integration is constrained by the three condition imposed above:

$$A_H = \int_0^{t_{max}} k_R \left(\frac{C'_H R}{k_H} \right) \left[1 - e^{-k_H t} \right] dt, \tag{9}$$

where

A_H = H-3 activity released via blowdown or evaporation (μCi), from time=0 to t_{max},
k_R = physical removal constant for either blowdown or evaporation (min^{-1}),
C'_H = initial H-3 concentration in either primary or the reflector (μCi/mL),
R = leak rate from either primary or the D_2O reflector into secondary (mL/min),
k_H = total H-3 removal constant, radioactive decay plus physical removal (min^{-1}), and
t_{max} = elapsed time (min); with a maximum value of 1440 *or* the time at which either of the conditions from Equations 7 or 8 are exceeded, whichever is the smallest.

Integrating Equation 9 yields:

$$A_H = k_R \left(\frac{C'_H R}{k_H} \right) \left[t + \frac{e^{-k_H t}}{k_H} \right]_0^{t_{max}} . \tag{10}$$

Substitute the following numerical values into Equation 10 to calculate A_H:

k_R = 1.85×10^{-3} min^{-1} (= blowdown rate of 4.41×10^4 mL/min divided by total secondary system water volume of 2.38×10^7 mL) *or*
4.23×10^{-3} min^{-1} (= evaporation rate of 1.01×10^5 mL/min divided by total secondary system water volume of 2.38×10^7 mL),
C'_H = 3.2×10^{-2} μCi/mL for primary or 4.15×10^3 μCi/mL for reflector,
R = arbitrarily chosen value; constrained by Equations 7 and 8,
k_H = 6.08×10^{-3} min^{-1}, and
t_{max} = arbitrarily chosen value; with a maximum value of 1440 min *or* the time at which either of the conditions from Equations 7 or 8 are exceeded, whichever is the smallest value.

Equation 10 yields an array of values for A_H, depending upon the numbers chosen for R and t_{max}. For each release scenario, the greatest value in the array represents the number that must be used for A_H (because the maximum value is the most conservative). For example, the maximum amount of H-3 released (by blowdown) for a leak via the D_2O reflector occurs when the leak rate from the reflector is 100 mL/min and the leak lasts for 605.6 min. Equation 8 is the condition that constrains the upper limit for A_H in this case, because if the product of the leak rate and the total leak time were any greater than 6056 mL, the total volume lost from the reflector would exceed the 16 gallon limit at which point the reactor operator would take corrective action to prevent further release.

The calculated values for H-3 activity released during a detected leak are summarized in Table 4.

In addition to the total activity released (A_H), the average tritium concentration of a release can be determined. During a leak from primary to secondary, the average H-3 concentration in the blowdown water is simply the total tritium activity released divided by the total volume of water discharged during the leak (A_H, the total activity released is $1.2x10^2$ μCi; the total water discharged via blowdown is constrained by Equation 8, and has a maximum value of $1.9x10^7$ mL). To calculate the tritium concentration in *air* resulting from evaporative discharge from the cooling towers during a leak from primary, first determine the average H-3 concentration of the evaporated water that is released. This will be the total tritium activity released via evaporation divided by the total volume of water discharged during the leak (A_H, the total activity released is $2.8x10^2$ μCi; the total water discharged via evaporation is constrained by Equation 8, and has a maximum value of $4.4x10^7$ mL).

Next, multiply the annual average water vapor content in Boston air, $6.8x10^{-6}$ mL H_2O / mL air (MITR-II Safety Analysis Report, Section 12.1.2.2), by the H-3 concentration of the evaporated water ($6.36x10^{-6}$ μCi/mL).

The calculated values for average tritium concentration of the release (for leaks via primary) are summarized in Table 4.

During a leak from the D_2O reflector to secondary, the average H-3 concentration in the water released via blowdown is simply the total tritium activity released divided by the total volume of water discharged during the leak (A_H, the total activity released is $5.6x10^7$ μCi; the total water discharged via blowdown is constrained by Equation 8, and has a maximum value of $2.7x10^7$ mL). To calculate the tritium concentration in *air* resulting from evaporative discharge from the cooling towers during a leak from the reflector, first determine the average H-3 concentration of the evaporated water that is released. This will be the total tritium activity released via evaporation divided by the total volume of water discharged during the leak (A_H, the total activity released is $1.3x10^8$ μCi; the total water discharged via evaporation is constrained by Equation 8, and has a maximum value of $6.1x10^7$ mL). Next, multiply the annual average water vapor content in Boston air, $6.8x10^{-6}$ mL H_2O / mL air (MITR-II Safety Analysis Report, Section 12.1.2.2), by the H-3 concentration of the evaporated water ($2.1x10^0$ μCi/mL).

The calculated values for average tritium concentration of the release (for leaks via the reflector) are summarized in Table 4. Clearly, these concentrations exceed the limits imposed by both the Technical Specifications and 10 CFR 20; however, the resulting doses to members of the general public will be negligible.

Table 4: H-3 Activity Released During a Detected Leak

Discharge Path	H-3 Activity Released	Concentration of Release (Average)
Blowdown (via Primary to Secondary)	$1.2x10^2$ μCi	$6.3x10^{-6}$ μCi/mL H_2O
Evaporation (via Primary to Secondary)	$2.8x10^2$ μCi	$4.3x10^{-11}$ μCi/mL air
Blowdown (via Reflector to Secondary)	$5.6x10^7$ μCi	$2.1x10^0$ μCi/mL H_2O
Evaporation (via Reflector to Secondary)	$1.3x10^8$ μCi	$1.4x10^{-5}$ μCi/mL air

Dose From H-3 Released During a Detected Leak:

Estimating the dose to a member of the public is straightforward for leaks from the primary system. Making the absolutely strictest conservative assumption that the entire H-3 release is taken up by a single individual, the maximum dose received is found by dividing the intake by the 10 CFR 20 ALI for H-3 (8×10^4 μCi for both ingestion and for inhalation). The calculated doses are in Table 5.

Following a tritium leak from the reflector, two additional steps are required to calculate the dose resulting from blowdown. The total tritium activity (2.1×10^0) is diluted by the water that is released during blowdown (2.7×10^7 mL), and it is further diluted by a factor of 20,000 at the ultimate point of discharge from the sewer system (from Section 3.8 of the Technical Specifications). Thus, the final H-3 concentration at discharge will be about 1.0×10^{-4} μCi/mL. This certainly represents a negligible dose impact to the public; an individual would have to take in over 150 liters of water at this concentration to receive a dose of about 1 mrem (based on the H-3 ALI of 8×10^4 μCi for ingestion).

The final consideration is a short-term discharge from the reflector to secondary, and the resulting evaporative release via the cooling towers. Taking no credit for plume dispersion, and assuming a stay time of limited duration within the plume, the dose to a member of the public will be:

$$D = \frac{(A_H/E)WC\ t_{stay}}{DAC\ \ 2000} 5000, \qquad (11)$$

where

- D = dose received (mrem) during a stay time of t_{stay} hours,
- A_H = H-3 activity released via evaporation (μCi),
- E = total volume of evaporated secondary water released from the cooling towers during discharge (mL),
- WC = annual average water vapor content in Boston air (6.8×10^{-6} mL H_2O / mL air),
- DAC = Derived Air Concentration for H-3 (μCi/mL), and
- t_{stay} = stay time for a member of the public in cooling tower evaporative plume (h).

Substitute the following numerical values into Equation 11 to calculate D:

- A_H = 1.3×10^8 μCi,
- E = 6.1×10^7 mL,
- t_{stay} = 2 h, and
- DAC = 2×10^{-5} μCi/mL.

Table 5: Max. Dose to a Member of the Public from H-3 Released During a Detected Leak

Leak Path	Dose to Public
Blowdown (via Primary to Secondary)	7.5×10^{0} mrem[a]
Evaporation (via Primary to Secondary)	1.8×10^{1} mrem[a]
Blowdown (via Reflector to Secondary)	9.8×10^{-1} mrem[b]
Evaporation (via Reflector to Secondary)	3.6×10^{0} mrem[c]

a: Dose (D) = total tritium activity released (A_H) / H-3 ALI
b: Dose (D) = (150 L of water consumed x tritium concentration at sewer discharge) / H-3 ALI
c: Dose calculated using Equation 11

Although the doses involved are small for a reflector leak, the total activity released (see Table 4) would be in excess of the limit in 10 CFR 20.2003(a)(4) of 5 Ci for H-3.

Na-24 Released During a Detected Leak:

The total Na-24 activity released (via blowdown) during a leak can be calculated by integrating an activity release rate. The lefthand side of Equation 12 contains an additional physical removal constant (k_R) that transforms the quantity being integrated from an atom release rate into an activity release rate.

The total Na-24 activity released during a leak will be as follows, with the caveat that the upper limit of integration is constrained by Equations 7 and 8:

$$A_{Na} = \int_0^{t_{max}} k_R \left(\frac{C'_{Na} R}{k_{Na}} \right) \left[1 - e^{-k_{Na} t} \right] dt, \tag{12}$$

where

A_{Na} = Na-24 activity released via blowdown (μCi), from time=0 to t_{max},
k_R = physical removal constant for blowdown (min^{-1}),
C'_{Na} = initial Na-24 concentration in either primary or the reflector (μCi/mL),
R = leak rate from either primary or the D_2O reflector into secondary (mL/min),
k_{Na} = total Na-24 removal constant, radioactive decay plus physical removal (min^{-1}), and
t_{max} = elapsed time (min); with a maximum value of 1440 *or* the time at which either of the conditions from Equations 7 or 8 are exceeded, whichever is the smallest.

Integrating Equation 12 yields:

$$A_{Na} = k_R \left(\frac{C'_{Na} R}{k_{Na}} \right) \left[t + \frac{e^{-k_{Na} t}}{k_{Na}} \right]_0^{t_{max}} . \tag{13}$$

Substitute the following numerical values into Equation 13 to calculate A_{Na}:

k_R = 1.85×10^{-3} min^{-1} (= blowdown rate of 4.41×10^4 mL/min divided by total secondary system water volume of 2.38×10^7 mL),
C'_{Na} = 5.18×10^{-1} µCi/mL for primary or 1.49×10^{-3} µCi/mL for reflector,
R = arbitrarily chosen value; constrained by Equations 7 and 8,
k_{Na} = 2.62×10^{-3} min^{-1}, and
t_{max} = arbitrarily chosen value; with a maximum value of 1440 min *or* the time at which either of the conditions from Equations 7 or 8 are exceeded, whichever is the smallest value.

Equation 13 yields an array of values for A_H, depending upon the numbers chosen for R and t_{max}. For each release scenario, the greatest value in the array represents the number that must be used for A_H (because the maximum value is the most conservative).

The values for Na-24 activity released during a detected leak are summarized in Table 6.

During a leak from primary to secondary, the average Na-24 concentration in the water released via blowdown is simply the total sodium activity released divided by the total volume of water discharged during the leak (the total activity released is 3.9×10^3 µCi; the total water discharged via blowdown is constrained by Equation 8, and has a maximum value of 1.9×10^7 mL).

During a leak from the reflector to secondary, the average Na-24 concentration in the water released via blowdown is simply the total tritium activity released divided by the total volume of water discharged during the leak (the total activity released is 3.2×10^1 µCi; the total water discharged via blowdown is constrained by Equation 8, and has a maximum value of 2.7×10^7 mL).

The calculated values for average Na-24concentration of the release are summarized in Table 6.

Table 6: Na-24 Activity Released During a Detected Leak

Discharge Path	Na-24 Activity Released	Concentration of Release (Average)
Blowdown (via Primary to Secondary)	3.9×10^3 µCi	2.0×10^{-4} µCi/mL H_2O
Blowdown (via Reflector to Secondary)	3.2×10^1 µCi	1.2×10^{-6} µCi/mL H_2O

Dose From Na-24 Released During a Detected Leak:

Estimating the dose to a member of the public is straightforward. Following a sodium leak from the primary system, the Na-24 is diluted by the water that is released during blowdown (1.9×10^7 mL), and it is further diluted by a factor of 20,000 at the ultimate point of discharge from the sewer system (from Section 3.8 of the Technical Specifications). Thus, the final Na-24 concentration at discharge will be about 1.0×10^{-8} µCi/mL. This certainly represents a negligible dose impact to the public, considering an individual would have to take in over 100,000 liters of water at this concentration to receive a dose of about 1 mrem (based on the Na-24 ALI of 4×10^3 µCi for ingestion). The calculation for a leak from the reflector is identical, except that the total

activity released is 3.2×10^{1} μCi, and the volume of water released during blowdown is 2.7×10^{7} mL. Again, there is a negligible dose impact to the public, considering an individual would have to take in over 1×10^{7} liters of water at this concentration to receive a dose of about 1 mrem (based on the Na-24 ALI of 4×10^{3} μCi for ingestion). The results are displayed in Table 7.

Table 7: Maximum Dose to a Member of the Public from Na-24 Released During a Detected Leak

Leak Path	Dose to Public
Blowdown (via Primary to Secondary)	1.3×10^{0} mrem[a]
Blowdown (via Reflector to Secondary)	7.4×10^{-1} mrem[b]

a: Dose (D) = (1E5 L of water consumed x Na-24 concentration at sewer discharge) / Na-24 ALI
b: Dose (D) = (1E7 L of water consumed x Na-24 concentration at sewer discharge) / Na-24 ALI

Conclusion:

In summary, using very conservative assumptions it can be seen that: for undetected leaks, the maximum H-3 activity released over one year via either blowdown or evaporation will fall within Technical Specification limits; and for higher rate short-term releases, although the activity discharged would exceed 5 Ci, the maximum dose to a member of the public would be less than 20 mrem.

For Na-24, the maximum concentrations in an undetected leak occurring over one year will be less than 10 CFR 20 limits; and for higher rate short-term releases, the maximum dose to a member of the public would only be about 1 mrem.

Appendix B

Additional Information for Response to Question 65

Dose Assessment from Airborne Activity within Contained Spaces of MITR-II

RESPONSE TO ITEM 65 OF THE SECOND PARTIAL REQUEST FOR ADDITIONAL INFORMATION

MASSACHUSETTS INSTITUTE OF TECHNOLOGY RESEARCH REACTOR DOCKET NO. 50-20

Section 11.1.1.4, Airborne Radiation Sources, Page 11-2. Please calculate the maximum anticipated annual dose to workers from airborne sources.

Response:

The principal source of radioactivity in generally accessible spaces within the containment building is ^{41}Ar. ^{41}Ar is generated through the neutron activation of stable argon which exists naturally as a small constituent of air. Although efforts to minimize the amount of air exposed to a neutron fluence have been made, there will be some air entrainment in systems and experimental facilities such that a small amount of ^{41}Ar will none-the-less be produced. Control of ^{41}Ar production is maintained through the use of cover gases and ventilation. For that quantity of ^{41}Ar that escapes into the containment atmosphere, the airborne activity concentration historically has been less than 10% of the derived air concentration (DAC).

^{41}Ar as a noble gas does not present itself as an inhalation hazard, but rather as an external dose component arising from submersion. The presumption of the derived air concentration for submersion values listed in 10CFR20 appendix B is that the submersion cloud is semi-infinite. A semi-infinite cloud is defined as a hemi-spherical cloud whose radius for photons is at least on the order of several mean free path lengths (μ^{-1}) in air. For ^{41}Ar, the mean free path length for the 1.29 MeV photon is approximately 140 meters. The radius for the containment building is 35 feet or 10.7 meters. Hence anywhere within the containment building a finite cloud condition would prevail. Approximation methods for the determination of the deep depth dose equivalent (DDE) from finite clouds have been generated. One method is that presented in U.S. Nuclear Regulatory Commission Regulatory Guide 1.183 (NRC2000). Specifically, equation 1 of section 4.2.7 of RG 1.183 provides an estimation of the DDE for finite clouds as follows:

$$DDE_{finite} = DDE_4 \text{ X } V^{0.338} / 1173 \qquad 1a$$

where:

DDE_{finite} is the deep depth dose from a finite cloud,
DDE_4 is the DDE from a semi-infinite cloud, and
V is the volume of a hemi-sphere of a finite cloud defined by a radius of R.

Rearranging equation 1a defines the fraction, f, of a dose from a finite cloud to that from a semi-infinite cloud as follows:

$$f = DDE_{finite}/DDE_4 = V^{0.338}/1173 \qquad 1b$$

An alternative method is that formulated by Skrable et.al. (Skrable 1972) where the dose from a finite cloud is determined as follows:

$$D_f \square D_4 f \qquad 2a$$

and

$$f = (\mu_{en} R) \qquad 2b$$

where:

D_f is the finite cloud dose,
D_4 is the semi-infinite cloud dose,
μ_{en} is the linear energy absorption coefficient (3.5E-3 m^{-1} for 1.29 MeV ^{41}Ar photons),
and R is the hemi-spherical cloud radius, m.

The DDE_4 (or D_4) may be determined from the DAC since by definition 2000 DAC-h represents a 5 rem committed dose.

In evaluating the value for the fraction of a semi-infinite cloud it should be observed that the dose to a receptor consists not only of the primary photon fluence but also of the scattered component (photon dose build-up factor) as well. The following table represents those spaces which may be considered accessible in the reactor building.

Table 1: Areas and Physical Descriptors

Area	Effective Length (feet)	Effective Width (feet)	Effective Height (feet)	Approximate Volume (ft^3)	Effective Radius (feet)	Effective Radius (meters)
Reactor Top	N/A	N/A	N/A	85750	35	10.7
Reactor Floor	68.2	25.6	40	69768	32	9.8
Fission Converter Medical Therapy Room	14.7	13.6	10	2000	9.9	3
Basement Medical Therapy Room	14.7	12.8	8	1510	8.9	2.7
Primary Chemistry	15.3	10.2	10	1569	9.1	2.8
Secondary Chemistry Area	13.6	9.1	10	1246	8.4	2.6
Control Room	20.5	11.9	8	1953	9.8	3
Basement Medical Room Set-up Area	23.8	11.9	8	2279	10.3	3.1
Equipment Room	34.1	23.9	10	8139	15.7	4.8

In determining the volume and effective radii, it is important to recognize that equipment and other obstructions would effectively reduce the effective radius and therefore the fraction of a semi-infinite cloud would be somewhat less.

Table 2 is a compilation of the effective radii and volumes from Table 1, the fractions of a semi-infinite cloud calculated via the two methods described above, and the effective DAC derived from the fraction of a semi-infinite cloud. The effective DAC is determined by the DAC listed in 10CFR20 Appendix B for ^{41}Ar (3.0 E-6) divided by the fraction of a semi-infinite cloud. Therefore, 2000 effective DAC-h would be equal to the primary limit of 5 rem DDE and the effective DDE rate for one DAC (10CFR20 Appendix B values) would be 2.5 mrem/h times the fraction of a semi-infinite cloud.

In considering the dose delivered to an individual it is important to recognize that the stay time in each area would vary and is significantly less than the assumed 2000 work hours per year. For example, the equipment room presents itself as a High Radiation Area during full power operations and therefore stay times are limited by procedure.

Table 2: Fraction of an Infinite Cloud and Effective DAC for ^{41}Ar in Various Spaces

Area	$R_{effective}$ (meters)	$V_{effective}$ (ft^3)	f Eq. 1b	f Eq. 2b	$DAC_{effective}$ Eq. 1b μCi/ml	$DAC_{effective}$ Eq. 2b μCi/ml
Reactor Top	10.7	85750	0.039	0.037	7.6E-5	8.0E-5
Reactor Floor	9.8	69768	0.037	0.034	8.1E-5	8.7E-5
Fission Converter Medical Therapy Room	3	2000	0.011	0.011	2.7E-4	2.8E-4
Basement Medical Therapy Room	2.7	1510	0.01	0.0096	2.9E-4	3.1E-4
Primary Chemistry	2.8	1569	0.010	0.0097	2.9E-4	3.1E-4
Secondary Chemistry Area	2.6	1246	0.0095	0.0089	3.2E-4	3.3E-4
Control Room	3	1953	0.011	0.0104	2.7E-4	2.9E-4
Basement Medical Room Set-up Area	3.1	2279	0.011	0.011	2.6E-4	2.7E-4
Equipment Room	4.8	8139	0.018	0.0168	1.7E-4	1.8E-4

The discussions presented thus far have focused on ^{41}Ar since this is the predominant source term with respect to airborne activity in normally occupied spaces. During periods of poor fuel performance (excessive off-gassing) trace quantities (i.e., much less than 1% of a DAC) of fission product gases (^{xx}Xe and ^{xx}Kr) had been detected. In reviewing the DACs listed for these fission

product gases, all are considered to be submersion based DACs including any short lived progeny (^{138}Cs and ^{88}Rb) arising from the decay of these radio-noble gases.

The controlling dose quantity of interest for the conditions presented above is for the external exposure. Personnel monitoring devices issued to all workers provides monitoring for these conditions. In the case of considering the shallow dose component under these conditions, the presumption of a finite cloud as defined for the photon component would not apply. The shallow dose component is more aptly defined for one half energy spatial equilibrium (2 ESE) conditions for which 2 ESE would prevail when the cloud dimensions are comparable to the range of the most energetic beta particle being emitted from the airborne contaminant. For the area/room dimensions reported here, the presumption of 2 ESE is appropriate and the airborne concentration as it relates to shallow dose may be used directly without modification. As stated earlier, the personnel monitoring devices will provide monitoring for these cases as well. In addition, the ratio of the shallow dose and the deep dose limit of 10 permits the control of all exposures based on the deep penetrating radiation component.

It should be recognized that these conditions prevail only while the reactor is operating, hence for maintenance activities, these sources of airborne activity no longer exist or are removed (decay and ventilation) shortly post shutdown. In the conduct of maintenance, refueling or other activities, where (particulate) airborne activity is possible to the extent of generating an intake requiring monitoring, engineering controls (ventilation, containment, decontamination, etc) are implemented. Airborne monitoring for these situations is also performed and may include general area air sampling and breathing zone air sampling as appropriate. Overall program effectiveness is based on fixed airborne monitoring stations, portable stations, breathing zone air sampling, contamination monitoring and bioassay monitoring as appropriate for the prevailing conditions. To date, intakes have been less than that required for combining internal and external exposures for total effective dose equivalent as required pursuant to '20.1202 and '20.1502(b) of 10CFR20.

Considering the above discussions, for routine operation the maximum stay-time expected for any individual would be the operator standing watch within the control room. This individual would be expected to have nominal stay-times of about 1000 hours per year. This value is estimated by considering that an individual work year is 2000 hours per year, periods of shutdown (capacity factor at ~70%) and periods of relief would effectively reduce this value by at least half. Therefore, for ^{41}Ar with a maximal concentration of 10% of a DAC and a fraction of a semi-infinite cloud for the control room at 0.011, the total number of DAC-h would be 1.1 DAC-h. This 1.1 DAC-h would correspond to approximately 2.75 mrem/y (based on 2000 DAC-h = 5000 mrem).

References:

(NRC 2000) U.S. Nuclear Regulatory Commission, *Alternative Radiological Source Terms for Evaluating Design Bases Accidents at Nuclear Power Reactors,* Regulatory Guide 1.183 (US NRC, Washington)

(Skrable 1972) "Evaluation of the Environmental Significance of the Projected 41Ar Release

from the Lowell Technological Institute Reactor", Health Physics 22(1), 49-56, Jan 1972, Kenneth W. Skrable, George Chabot, Joseph Killelea, Harold Wedlick

Appendix C

Additional Information for Response to Question 66

RESPONSE TO ITEM 66 OF THE SECOND PARTIAL REQUEST FOR ADDITIONAL INFORMATION

MASSACHUSETTS INSTITUTE OF TECHNOLOGY RESEARCH REACTOR DOCKET NO. 50-20

Section 13.2.1.3, Dilution Factor, @ Page 11-24. Please provide details of your calculations that provide the basis of the dilution factor of 50,000 and the dose scaling factor of 1,200. What would be the dose from routine annual release of effluent (assuming continuous 6MW operation) to the maximum exposed member of the public and at the nearest residence?

Response:

The Massachusetts Institute of Technology's Research Reactor has been using the concept of dilution factors for determining compliance with gaseous effluents as is specified within the current Technical Specifications for MITR-II. Specifically, the dilution factor is used to predict the environmental concentration from a stack concentration by dividing the stack concentration by the dilution factor. This permits a direct comparison to the 10CFR20 Appendix B Effluent Concentration values as permitted within 10CFR20.1301 and 20.1302. Within annual reports to the NRC, the effluent concentration, total activity discharged and the percentage of the Technical Specifications they represent is reported (i.e., 10CFR20 Appendix B values and using the permitted dilution factor). The current MITR-II Technical Specifications allows for a dilution factor of 3000 and a re-concentration factor of 700 for radionuclides greater that 8 days. The re-concentration factor was presumably developed for ground deposition activity resulting in a pathway other than inhalation and submersion. In addition to the requirements of 10CFR20.1301 and 1302, the NRC and the EPA had entered into a memorandum of understanding that resulted in a lower constraint limit for members of the general public. As defined in 10CFR20.1101, the new constraint limit is 10 mrem/y (Total Effective Dose Equivalent).

This application for renewal has maintained the concept of dilution factors in order to maintain continuity in approach with our present and established methodologies. The MITR-II technical specifications for the permitted dilution factor have always been recognized as being overly conservative, which in general had not been an issue. However, it does not permit realistic consideration for offsite dose consequences. This became apparent in the adoption of the EPA limit of 10 mrem EDE as a constraint limit within 10CFR20 where the actual doses calculated from a release which meet the constraint limit would appear to be at odds with the reported percentages of the Technical Specifications in the annual reports. For example, the Effective Dose Equivalent for determining compliance with the constraint limit has consistently been calculated to be approximately 1 mrem/y using the U.S. Environmental Protection Agency's

Comply Code, whereas the percentage of technical specifications as reported in the annual report has been on the order of 60% for ^{41}Ar. In as much as 10CFR20 Appendix B values of effluent concentration limits are based on a dose of 50 mrem/y, a dose of 30 mrem/y could be inferred from the reported percentage of Technical Specification which would be at odds with the constraint limit determination of 1 mrem/y if not interpreted correctly. This is further apparent when comparing the environmental monitoring data (as reported in the Annual report) which is consistent with the constraint limit determinations

It was therefore appropriate to consider in this application more realistic values for a Dilution Factor for routine operation of this facility. The dilution factor, DF is the ratio of the stack concentration as compared to the expected concentration in the environment and as follows:

$$DF / \chi_s/\chi_{DW} \qquad 1$$

Where:

DF is the Dilution Factor (unitless),
χ_s is the stack concentration ($\mu Ci/cm^3$), and
χ_{DW} is the down wind concentration at a point of interest ($\mu Ci/cm^3$).

The concentration downwind, χ_{DW}, can be obtained from the atomospheric dispersion parameter, χ_{DW}/Q_s (s/cm^3), multiplied by the stack release rate, Q_s ($\mu Ci/s$) as follows:

$$\chi_{DW} = (\chi_{DW}/Q_s) * Q_s \qquad 2$$

The concentration in the stack may be determined from the activity release rate and the stack flow rate as follows:

$$\chi_S = Q_s * F_s \qquad 3$$

Where, Fs is the stack flow rate (cm^3/s) and all other terms are as previously defined.

Substituting equations 3 and 2 into equation 1, the resulting expression is obtained.

$$DF = F_s / (\chi_{DW}/Q_s) \qquad 4$$

The atmospheric dispersion parameter (χ_{DW}/Q_s) may be determined by a number of methods. The method most commonly used and accepted is referred to as the straight line gaussian dispersion model/equation. It is this method that is used in the EPA comply code, the EPA CAP88-PC code, the NRC XOQ/DOQ code and a host of others. The output of the later two codes presents tabular values of the atmospheric dispersion parameter as a function of distance, direction and atmospheric stability class. The computer code CAP88-PC ver 2.10 was used to obtain the tabulated values of the annual average χ_{DW}/Q_S for an elevated release in an open site.

The equations used are consistent with the methods used in US NRC regulatory Guides 1.109 and 1.111. Details on the application of the straight line gaussian plume model, are found in Regulatory Guides 1.109, 1.111 and the CAP88PC user manual.

The wind profile information used for this input was the stability array (STAR) data obtained from the National Climatic Data Center (NCDC) for the Boston area and were included in the model by the author of CAP88PC at the request of the reactor facility. A wind rose plot of the NCDC data is provided within Figure 1 for Boston's Logan airport for the years 1984 through 1992.

The radionuclide chosen for gaseous effluents in determining the DF was ^{41}Ar since this is the predominant radionuclide produced and released. ^{41}Ar is produced and released at a near constant rate of about 1.2 ± 0.2 Ci/MWD (average of 13 years). In comparison, fission product gases (noble gases) represent approximately 1% of the total activity released.

The maximal value of χ_{DW}/Q_S calculated for noble gases (^{41}Ar in this case) is about 4 E-7 at a distance of 900 meters ESE (represented as the wind direction toward ESE). The values of χ_{DW}/Q_S are shown in Table 1 along with a graphical depiction in Figure 2.

A nominal stack flow rate of 8440 $ft^3\ min^{-1}$ (3.983 $m^3\ s^{-1}$) and χ_{DW}/Q_S results in a calculated Dilution Factor of 9.7 E6 (~ 1 E7). In the original submittal of the SAR for re-licensing, a Dilution Factor of 50,000 was proposed. After review by the MIT Committee on Reactor Safeguards of the information provided herein, it was recommended that a Dilution Factor of 100,000, a two order of magnitude below the calculated value be applied.

In determining the dose (rate) from a release, the following equation is applicable:

$$D = (\chi_{DW}/Q_S)*Q_S*DCF$$

Where:

D is the dose in mrem/y,
DCF is the Dose Conversion Factor for submersion in a semi-infinite cloud (mrem/y per uCi/cm^3), and
all other terms are as previously defined.

Alternatively, in the application of a Dilution Factor the above equation may be rewritten as follows:

$$D = (\chi_S/DF) * DCF.$$

Continuous operation at 6 MW would result in a power generation 2190 MWD/y. Applying the average activity released per unit power of 1.2 Ci/MWD, the resulting activity released per year

would be nominally 2628 Ci. A dose conversion factor from EPA 402-R-93-081, Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil"(FGR12) was obtained for ^{41}Ar. The value listed is 6.50E-14 Sv Bq^{-1} m^3 s^{-1}, which in traditional units would be 7.59E-3 mrem pCi^{-1} m^3 y^{-1}. This later value can be compared to the value listed in Regulatory Guide 1.109 (RG-1.109)Table B-1 for ^{41}Ar at 8.84 E-3.

In using these values the maximum submersion dose would be estimated as 1.59 mrem/y using DF of 100,000 and DCF from FGR12 and 1.85 mrem/y using the same DF and the DCF from RG-1.109. In using the calculated χ/Q, then the estimated dose using FGR12 and RG-1.109 would be 1.63 E-2 and 1.90 E-2 respectively.

The nearest resident is determined to be a dormitory located at approximately 100m from the center of the containment building to the corner of the dormitory residence. The distance from the stack to the dorm is not significantly different and will be assumed to be the same. In this regard, the values of χ/Q are much smaller at 2.4 E-9 s m^{-3} (nearly two orders of magnitude). Under these conditions (distances), the principal exposure is not from submersion dose but from direct exposure from a plume aloft. The methodology presented within Regulatory Guide 1.109 for a finite-cloud gamma χ/Qs using sector averaging is not applicable at distances close to the source. An analysis of the radiological impact on a proposed structure to be located 60 meters from the reactor exhaust stack had been conducted, accounting for the elevated plume condition and re-entrainment of the exhaust into the building. One of the analyses (preliminary evaluation) modeled the elevated plume as a line source. The result indicated a maximum exposure of 1.12 mrem/y based on an average elevation 35 meters with a frequency of wind in that direction of 10%, assuming 100% occupancy, and not accounting for shielding offered by the structure (i.e., free-in-air receptor). These calculations were performed based on an annual average discharge of 1500 Ci. Scaling this linearly to 2628 Ci for continuous operation at 6MW, the estimated exposure from a plume aloft would be approximately 2 mrem/y. Environmental monitors are located throughout campus, one of which is located on the roof top of this dormitory. Historical readings (7 year average) for this dormitory have resulted in an average dose of 0.313±0.133 mrem/y. Again assuming a linear relationship, a 6MW continuous operation may realize an exposure on the order of 0.6 mrem/y.

The current MITR-II Technical Specifications allows for a Dilution factor of 3000 and a re-concentration factor of 700 for radionuclide particulates with a half life greater that eight (8) days. The re-concentration factor was presumably developed for ground deposition activity and bio-accumulation resulting in a pathway other than inhalation and submersion. Specifically, within the current SAR for MITR-II it is stated that Afor a re-concentration ratio of 1000 as discussed in the proposed Appendix I to 10CFR50. The current Appendix I does not discuss re-concentration factors, rather Appendix I uses a dose based methodology for ensuring design criteria doses are maintained ALARA. Guidance for implementing Appendix I (Regulatory Guide 1.109) does discuss bio-accumulation through various pathways.

In as much as this facility is equipped with HEPA filters, the only particulate radionuclide with a half life greater than eight days that is considered would be ^{131}I. In reviewing Regulatory Guide 1.109, for the Stable Element Transfer Data for Iodine (Table E-1) a summation of these values would yield a transfer value 561.5. Coupled with ground deposition exposure pathways, the original value of 700 within the current Technical Specifications would appear appropriate. Therefore a conservative value of 1000 would likewise be appropriate and not unduly conservative.

In summary, using the dilution factor of 100,000 as amended is more realistic than previous dilution factors used in MITR-II. The re-concentration factor previously used is also appropriate, however, a conservative value of 1000 is requested. Projected doses from continuous operation are well below the regulatory limit using the proposed values and well below the ALARA constraint limit.

Figure 1

Wind Rose Plot from NCDC Data for Boston's Logan Airport for the years 1984 through 1992

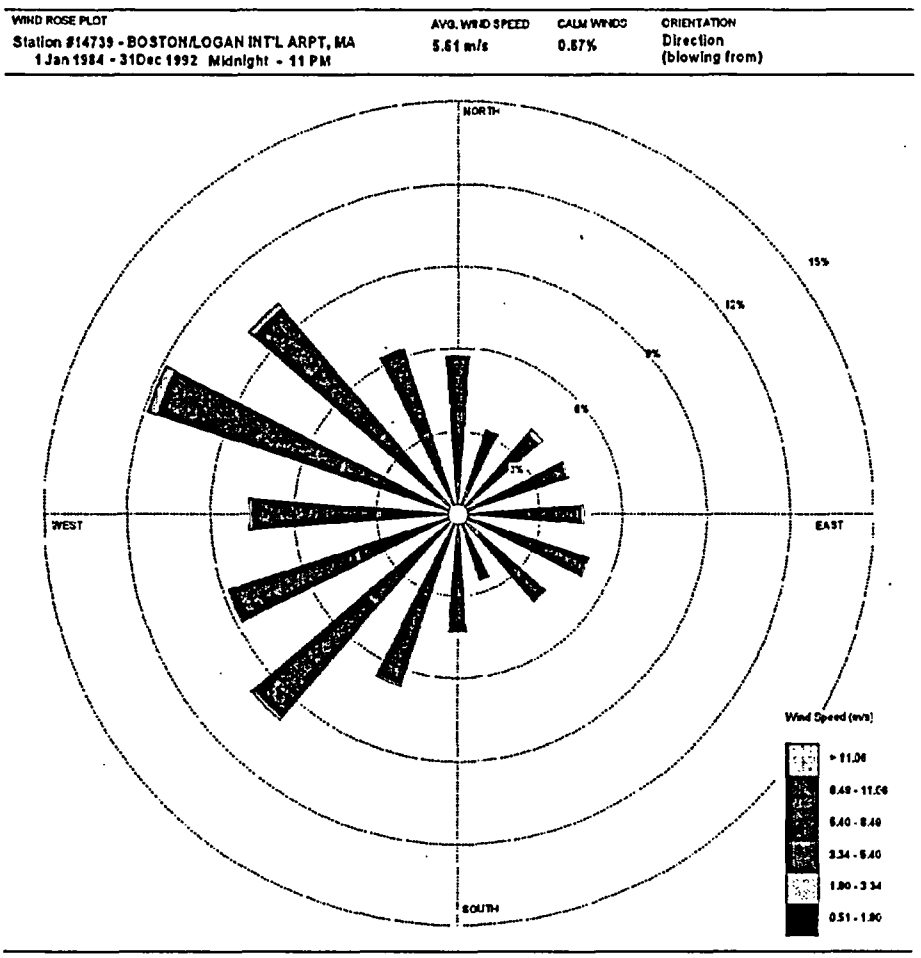


Table 1

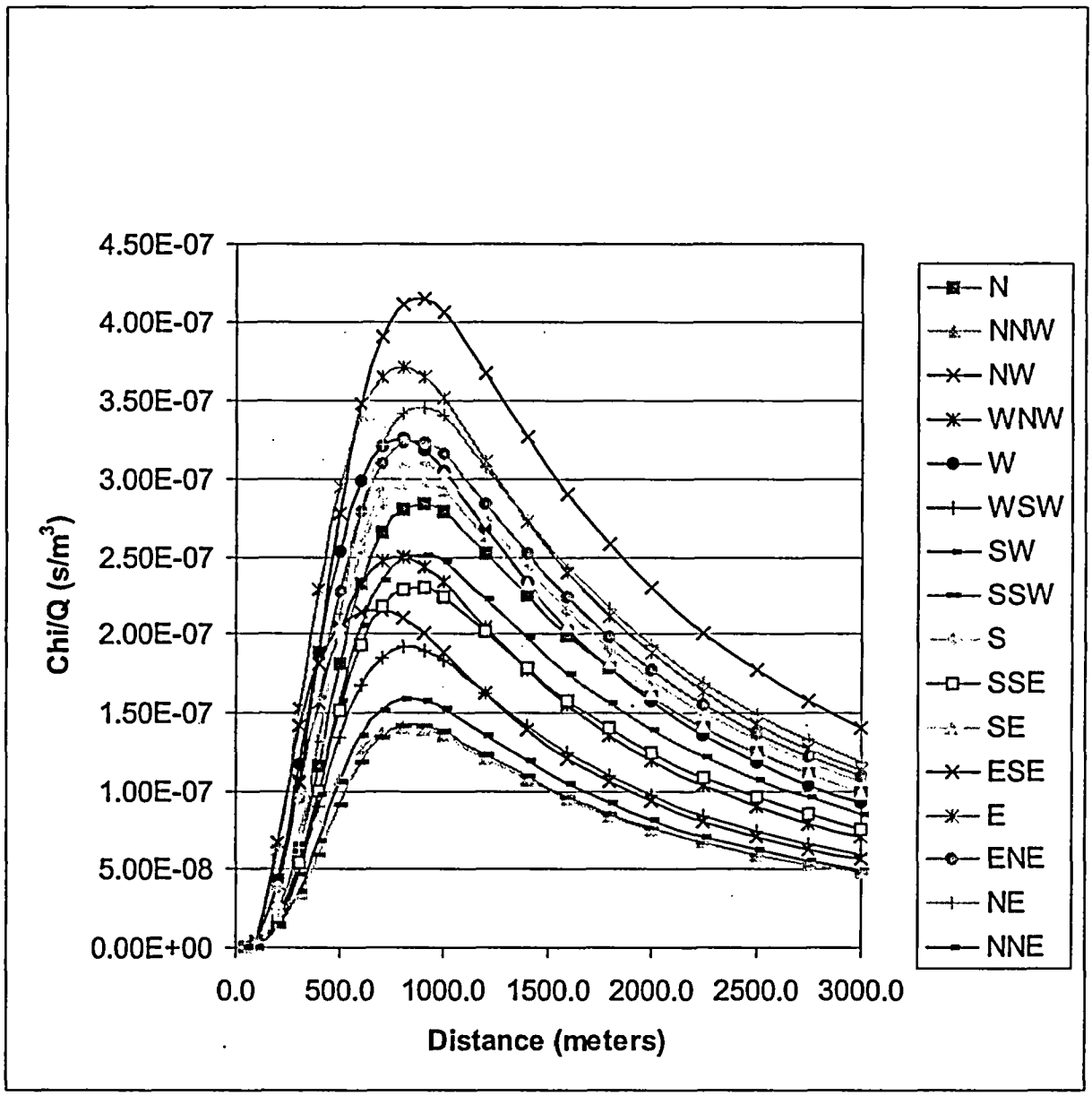
Ground Level Π/Q (s m^{-3}) for ^{41}Ar at Various Distances and 16 Wind Directions

Wind Dir.	Downwind Distance (meters)									
	50.0	100.0	200.0	300.0	400.0	500.0	600.0	700.0	800.0	900.0
N	2.1E-20	1.8E-10	2.5E-08	6.3E-08	1.2E-07	1.8E-07	2.3E-07	2.7E-07	2.8E-07	2.8E-07
NNW	2.5E-13	5.5E-10	2.0E-08	5.2E-08	8.1E-08	1.1E-07	1.3E-07	1.4E-07	1.4E-07	1.4E-07
NW	7.0E-13	3.3E-09	6.7E-08	1.4E-07	1.8E-07	2.0E-07	2.1E-07	2.2E-07	2.1E-07	2.0E-07
WNW	6.0E-13	1.3E-09	4.3E-08	1.1E-07	1.6E-07	2.0E-07	2.3E-07	2.5E-07	2.5E-07	2.4E-07
W	6.3E-13	2.3E-09	4.3E-08	1.2E-07	1.9E-07	2.5E-07	3.0E-07	3.2E-07	3.3E-07	3.2E-07
WSW	4.8E-14	7.6E-10	1.8E-08	5.0E-08	9.1E-08	1.3E-07	1.7E-07	1.9E-07	1.9E-07	1.9E-07
SW	2.5E-13	5.2E-10	1.4E-08	3.6E-08	6.8E-08	1.1E-07	1.4E-07	1.5E-07	1.6E-07	1.6E-07
SSW	8.1E-21	9.5E-11	1.3E-08	3.2E-08	5.9E-08	9.2E-08	1.2E-07	1.3E-07	1.4E-07	1.4E-07
S	2.7E-20	1.8E-10	2.5E-08	7.2E-08	1.3E-07	2.0E-07	2.6E-07	2.9E-07	3.1E-07	3.1E-07
SSE	2.8E-20	1.5E-10	2.0E-08	5.4E-08	1.0E-07	1.5E-07	1.9E-07	2.2E-07	2.3E-07	2.3E-07
SE	2.0E-13	5.1E-10	2.6E-08	7.4E-08	1.3E-07	2.0E-07	2.5E-07	2.8E-07	3.0E-07	3.0E-07
ESE	1.2E-12	2.4E-09	4.1E-08	1.1E-07	1.9E-07	2.8E-07	3.5E-07	3.9E-07	4.1E-07	4.1E-07
E	6.0E-13	2.3E-09	6.2E-08	1.5E-07	2.3E-07	2.9E-07	3.4E-07	3.6E-07	3.7E-07	3.6E-07
ENE	1.1E-12	2.2E-09	3.8E-08	9.6E-08	1.6E-07	2.3E-07	2.8E-07	3.1E-07	3.2E-07	3.2E-07
NE	2.3E-20	1.5E-10	2.1E-08	6.3E-08	1.3E-07	2.1E-07	2.8E-07	3.2E-07	3.4E-07	3.5E-07
NNE	1.4E-20	8.3E-11	1.3E-08	4.6E-08	9.8E-08	1.6E-07	2.1E-07	2.4E-07	2.5E-07	2.5E-07

Wind Dir.	Downwind Distance (meters)									
	1000.0	1200.0	1400.0	1600.0	1800.0	2000.0	2250.0	2500.0	2750.0	3000.0
N	2.8E-07	2.5E-07	2.2E-07	2.0E-07	1.8E-07	1.6E-07	1.4E-07	1.2E-07	1.1E-07	1.0E-07
NNW	1.4E-07	1.2E-07	1.1E-07	9.5E-08	8.5E-08	7.6E-08	6.7E-08	5.9E-08	5.3E-08	4.8E-08
NW	1.9E-07	1.6E-07	1.4E-07	1.2E-07	1.1E-07	9.4E-08	8.1E-08	7.1E-08	6.3E-08	5.6E-08
WNW	2.3E-07	2.0E-07	1.8E-07	1.6E-07	1.4E-07	1.2E-07	1.0E-07	9.1E-08	8.0E-08	7.1E-08
W	3.1E-07	2.7E-07	2.3E-07	2.0E-07	1.8E-07	1.6E-07	1.4E-07	1.2E-07	1.0E-07	9.3E-08
WSW	1.8E-07	1.6E-07	1.4E-07	1.2E-07	1.1E-07	9.8E-08	8.5E-08	7.4E-08	6.6E-08	5.8E-08
SW	1.5E-07	1.4E-07	1.2E-07	1.1E-07	9.3E-08	8.2E-08	7.1E-08	6.3E-08	5.5E-08	4.9E-08
SSW	1.4E-07	1.2E-07	1.1E-07	9.7E-08	8.6E-08	7.7E-08	6.7E-08	5.9E-08	5.3E-08	4.7E-08
S	3.0E-07	2.7E-07	2.4E-07	2.1E-07	1.9E-07	1.7E-07	1.5E-07	1.3E-07	1.2E-07	1.1E-07
SSE	2.2E-07	2.0E-07	1.8E-07	1.6E-07	1.4E-07	1.3E-07	1.1E-07	9.6E-08	8.5E-08	7.6E-08
SE	2.9E-07	2.6E-07	2.3E-07	2.1E-07	1.8E-07	1.6E-07	1.4E-07	1.2E-07	1.1E-07	9.9E-08
ESE	4.1E-07	3.7E-07	3.3E-07	2.9E-07	2.6E-07	2.3E-07	2.0E-07	1.8E-07	1.6E-07	1.4E-07
E	3.5E-07	3.1E-07	2.7E-07	2.4E-07	2.1E-07	1.9E-07	1.6E-07	1.4E-07	1.3E-07	1.1E-07
ENE	3.2E-07	2.8E-07	2.5E-07	2.2E-07	2.0E-07	1.8E-07	1.6E-07	1.4E-07	1.2E-07	1.1E-07
NE	3.4E-07	3.1E-07	2.7E-07	2.4E-07	2.2E-07	1.9E-07	1.7E-07	1.5E-07	1.3E-07	1.2E-07
NNE	2.5E-07	2.2E-07	2.0E-07	1.8E-07	1.6E-07	1.4E-07	1.2E-07	1.1E-07	9.6E-08	8.6E-08

Figure 2

Ground Level Π_{DW}/Q_S (^{41}Ar) as a Function of Distance for Various Directions (towards)



References:

1. (NRC 1977a) U.S. Nuclear Regulatory Commission, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I* Regulatory Guide 1.109 Rev. 1, October 1977, (USNRC, Washington).

2. (NRC 1977b) U.S. Nuclear Regulatory Commission, *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors*, Regulatory Guide 1.111, Rev. 1, July 1977, (USNRC, Washington).

3. (EPA-web) U.S. Environmental Protection Agency (EPA) "Support Center for Regulatory Air Models" (SCRAM) World Wide Web site, www.epa.gov/scram001.

4. (EPA-1988) U.S. Environmental Protection Agency, *Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,* Federal Guidance Report No. 11, EPA 520/1-88-020, (ORNL, September 1988).

5. (EPA-1993) U.S. Environmental Protection Agency, *External Exposure to Radionuclides in Air, Water, and Soil*, Federal Guidance Report No.12, EPA 402-R-93-081 (ORNL, September 1993).

Appendix D

Additional Information for Response to Question 92(b)

File Memo dated 29 April 03, "Loss of Primary Flow Transient Analysis"

NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
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Activation Analysis
Coolant Chemistry
Nuclear Medicine
Reactor Engineering

MEMORANDUM

TO: MITR Files

FROM: Lin-Wen Hu

DATE: April 29, 2003

RE: Loss of Primary Flow Transient Analysis (2)

1. The loss of primary flow transient analysis was originally performed using initial conditions of reactor power 6.1 MW, primary flow 2000 gpm, coolant outlet temperature of 55 °C, and coolant height at 10 ft (LOF case#1). This analysis was repeated using the LSSS as the initial conditions (LOF case#2). The LSSS for the MITR-III are: reactor power 7.4 MW, primary flow 1800 gpm, coolant outlet temperature 60 °C, and coolant height at 10 ft. The MULCH-II code was used for both analyses. All other assumptions are the same for both analyses.

2. Figures 1 and 2 are comparisons of the coolant outlet temperatures of the average and hot channels for the two cases. Note that the initial coolant temperatures are higher in Figure 2 because of the higher initial power (7.4 MW v.s. 6.1 MW) and lower initial flow rate (1800 gpm v.s. 2000 gpm). The peak hot channel outlet coolant temperatures, which occur around 1.5 s into the transient, are 105.2 °C for case#2 and 97.0 °C for case#1. Note that the coolant temperature then decreases rapidly in both cases because of reactor scram. Both analyses showed that the hot channel coolant outlet temperature would reach saturation after about 15 to 20 seconds. Figure 3 shows the calculated fuel temperatures at the average and hot channel outlet assuming the initial conditions of the LOF transient are LSSS. The calculated fuel temperatures are well below the cladding softening point of 450 °C.

3. Figure 4 is the calculated reactor decay power assuming equilibrium reactor power was at 7.4 MW before scram. The reactor decay heat at 16 seconds after reactor scram is about 325 kW. As shown in SAR section 4.6.6.3, the best-estimate dry-out condition is 468 kW.

4. The MULCH-II output file for LOF case#2 is attached to this memo.

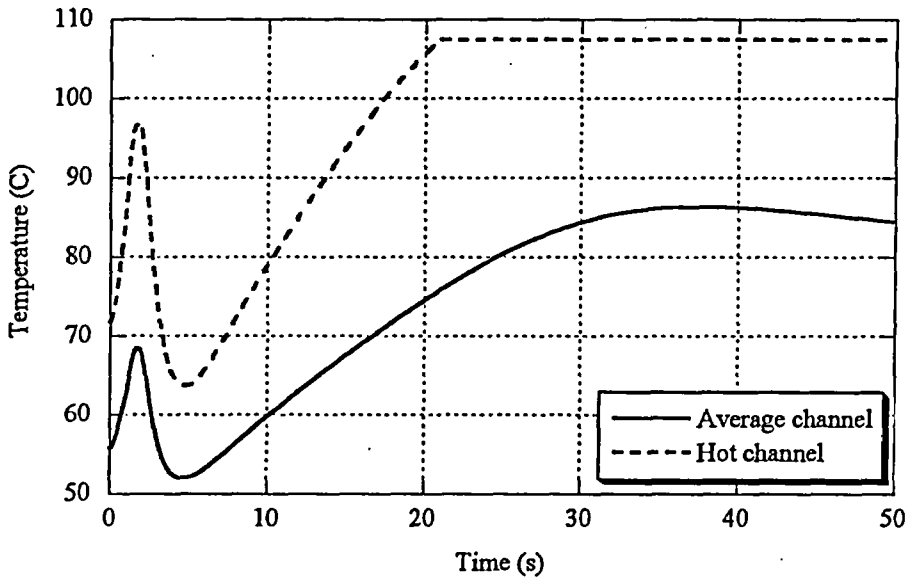

Figure 1. Coolant outlet temperatures of average and hot channels during a loss of primary flow transient. The initial conditions used for this analysis are reactor power at 6.1 MW, primary flow 2000 gpm, coolant outlet temperature 55 °C, and coolant height at 10 ft.

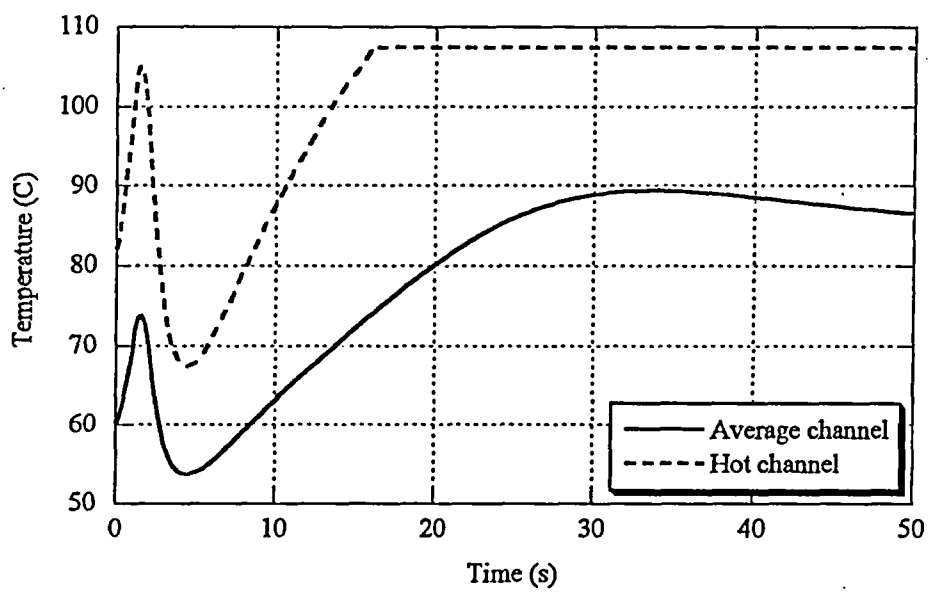

Figure 2. Coolant outlet temperatures of average and hot channels during a loss of primary flow transient. The initial conditions used for this analysis are reactor power at 7.4 MW, primary flow 1800 gpm, coolant outlet temperature 60 °C, and coolant height at 10 ft.

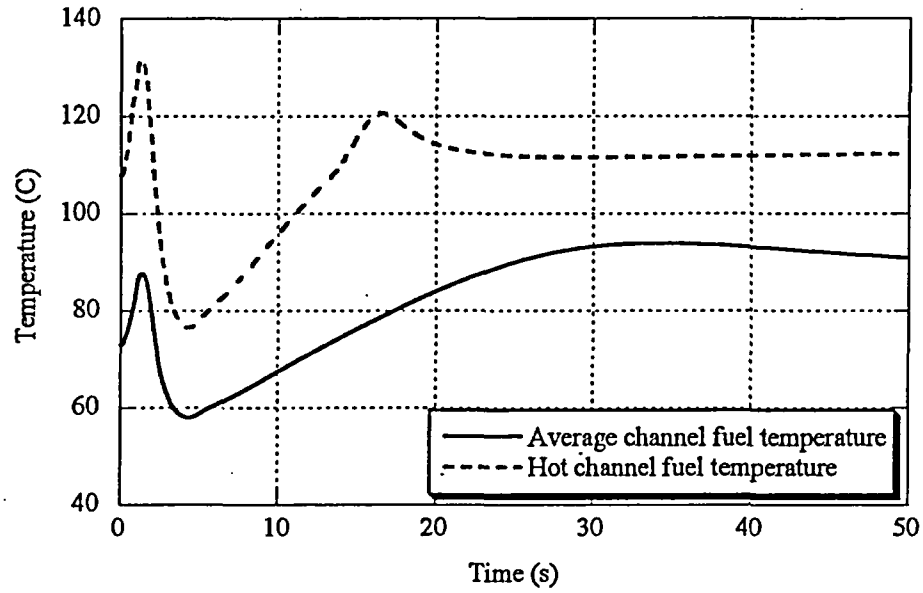

Figure 3. Fuel temperatures of average and hot channels at outlet during a loss of primary flow transient. The initial conditions used for this analysis are reactor power at 7.4 MW, primary flow 1800 gpm, coolant outlet temperature 60 °C, and coolant height at 10 ft.

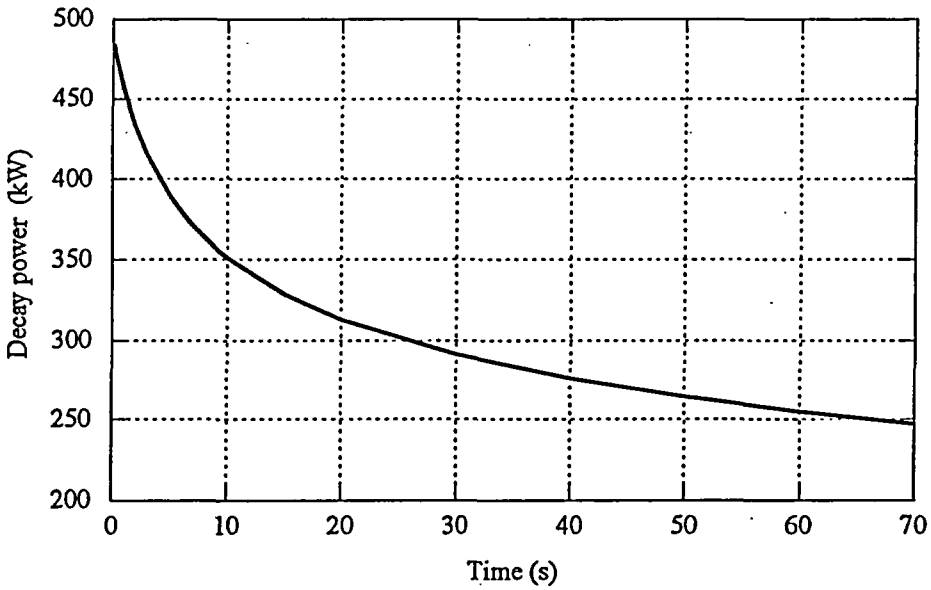

Figure 4 Reactor decay power calculated using DKPOWR assuming equilibrium power of 7.4 MW before reactor scram.

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-----------------------------------------------------
      Multi-Channel Analysis Code, MULCH-II
    MIT Nuclear Reactor Laboratory  7/15/1996
-----------------------------------------------------
```

LOSS OF FLOW PREDICTION FOR MITR-III best estimate

Reactor Power (kW)= 7400.00 Cooling Tower Outlet Temp (C)= 13.00
Primary Flow (kg)= 111.00 Secondary Flow (kg)= 103.00 Cooling Tower Efficiency= .80
Reference Temp (C)= 50.00 Coolant height from air/water interface to top of flow guide (m)= 2.31

*** simulated case is --> LOSS OF PRIMARY FLOW ***
Steady-State Operation before Shutdown for ****** hours
Time Step (s)=.100E+00 Total Simulation Time (s)= 50.00
Instrument Delay Time (s)= 1.00 80% Blade Insertion Time (s)= 1.00

Pump Coastdown Curve:
(exp(-1.870+ .410*t/10+ 2.950*exp(t/10)+ -.680*exp(-(t/10)^2))- .514)/(1.492- .514)

Loop Component Geometries:

I	Aflow(m^2)	Vol(m^3)	De(m)	dz(m)	Kform	Nchan
1	.320E-01	.427E+00	.203E+00	-7.08	4.58	1
2	.389E-04	.168E-03	.704E-02	.00	7.30	1770
3	.320E-01	.468E+00	.203E+00	6.97	2.17	1
4	.339E+00	.413E+00	.180E+00	-1.22	.00	1
5	.111E+00	.760E-01	.630E-01	-.69	.30	1
6	.440E-02	.160E-01	.220E+00	-.01	.18	1
7	.290E-01	.180E-01	.400E-01	-.61	.00	1
8	.125E-03	.824E-04	.219E-02	.66	2.05	345
9	.130E+00	.990E-01	.387E+00	.76	.00	1
10	.923E+00	.192E+01	.108E+01	1.22	.00	1
11	.320E-01	.427E+00	.203E+00	-7.08	4.58	1
12	.900E-04	.389E-03	.301E-02	.00	7.30	1770
13	.320E-01	.468E+00	.203E+00	6.97	2.17	1

Anti-siphon and Natural Convection Valve Geometries:

	Acont(m^2)	Aref(m^2)	Vball(m^3)	Rball(kg/m^3)	Kup	Kdown	Nv
ASV	.178E-02	.384E-02	.1059E-03	2715.00	7.90	6.90	2
NCV	.271E-02	.811E-02	.2040E-03	2715.00	******	******	4

[redacted]

Fraction of coolant cooling the fueled region= .920
HX Fouling factor (C m^2/W)= .3500E-03

Fraction of energy deposited in fuel= .910 Coolant= .054 D2O= .021 Graphite= .015
Hot Channel Factor= 2.000

	bottom 2	3	4	5	6	7	8	9	10	top 11
Shape_avg	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
Shape_hot	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
Peak_avg	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
Peak_hot	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000

Minimum flow distribution in flow channel= .8640

Engineering Factors for:
Reactor Power= 1.000 Hot Channel Flow Rate= 1.000
Heat Transfer Coef= 1.000 Hot Spot Heat flux= 1.000

Min CHF ratio= 1.500 Min DNB ratio= 1.500

---------- END OF INPUT ----------

---------- START OF OUTPUT ----------
DP_core= 41528.180000
DPratio1= -2.104542E-01

** steady-state temperatures for each components **
1 2 3 4 5 6 7 8 9 10 11 12 13

59.0 43.7 43.7 43.7 43.7 43.7 43.7 59.0 59.0 59.0 13.0 29.5 29.5

** steady-state temperatures for core region **
Tw_hot=coolant temperature at the hot channel
Tw_avg=coolant temperature at the average channel
Tc_hot=clad temperature at the hot channel
Tc_avg=clad temperature at the average channel
Tf_hot=fuel temperature at the hot channel
Tf_avg=fuel temperature at the average channel

	1	2	3	4	5	6	7	8	9	10	11	12
Tw_hot	43.7	47.5	51.4	55.2	59.1	63.0	66.8	70.7	74.5	78.4	82.2	82.2
Tw_avg	43.7	45.3	47.0	48.7	50.3	52.0	53.6	55.3	57.0	58.6	60.3	60.3
Tc_hot	43.8	68.7	71.8	75.1	78.2	81.6	85.1	88.6	92.2	95.8	99.4	82.4
Tc_avg	43.8	55.2	56.7	58.2	59.7	61.3	62.9	64.4	65.9	67.4	69.0	60.4
Tf_hot	43.7	77.0	80.2	83.4	86.8	90.1	93.6	97.1	100.7	104.2	107.9	82.2
Tf_avg	43.7	59.1	60.6	62.1	63.6	65.2	66.7	68.2	69.8	71.4	72.9	60.3

Qflux_h .1000E-04 .3234E+06 .3234E+06 .3234E+06 .3234E+06 .3234E+06 .3234E+06 .3234E+06 .3234E+06 .3234E+06 .3234E+06 .1000E-04

** Cladding Temperature at ONB and CHFR **

	1	2	3	4	5	6	7	8	9	10	11	12
TONB_hot	107.5	114.3	114.3	114.3	114.3	114.3	114.3	114.3	114.3	114.3	114.3	107.5
TONB_avg	107.5	112.4	112.4	112.4	112.4	112.4	112.4	112.4	112.4	112.4	112.4	107.5
CHFR	999.9	9.1	9.1	9.1	9.1	9.1	9.1	9.1	9.1	9.1	9.1	999.9

LSSS OK!
Safety Limits OK!

** LSSS and Safety Limits Index **
0: below limit 1: limit exceeded

	1	2	3	4	5	6	7	8	9	10	11	12
LSSS	0	0	0	0	0	0	0	0	0	0	0	0
SL	0	0	0	0	0	0	0	0	0	0	0	0

0.000000E+00	102.175500	0.000000E+00	0.000000E+00

Scram signal sent at: 1.000000E-01(s)
Blades 80% inserted at: 2.100000(s)

1.000000E-01	98.917020	0.000000E+00	0.000000E+00
2.000000E-01	92.767810	0.000000E+00	0.000000E+00
3.000000E-01	87.031840	0.000000E+00	0.000000E+00
4.000000E-01	81.677700	0.000000E+00	0.000000E+00
5.000000E-01	76.676730	0.000000E+00	0.000000E+00
6.000000E-01	72.002690	0.000000E+00	0.000000E+00
7.000000E-01	67.631540	0.000000E+00	0.000000E+00
8.000001E-01	63.541280	0.000000E+00	0.000000E+00
9.000001E-01	59.711720	0.000000E+00	0.000000E+00
1.000000	56.124330	0.000000E+00	0.000000E+00
1.100000	52.762060	0.000000E+00	0.000000E+00
1.200000	49.609260	0.000000E+00	0.000000E+00
1.300000	46.651530	0.000000E+00	0.000000E+00
1.400000	43.875560	0.000000E+00	0.000000E+00
1.500000	41.269100	0.000000E+00	0.000000E+00
1.600000	38.820880	0.000000E+00	0.000000E+00
1.700000	36.520440	0.000000E+00	0.000000E+00
1.800000	34.358120	0.000000E+00	0.000000E+00
1.900000	32.325000	0.000000E+00	0.000000E+00
2.000000	30.412800	0.000000E+00	0.000000E+00
2.100000	28.613860	0.000000E+00	0.000000E+00
2.200000	26.921060	0.000000E+00	0.000000E+00
2.300000	25.327790	0.000000E+00	0.000000E+00
2.400000	23.827910	0.000000E+00	0.000000E+00
2.500000	22.415700	0.000000E+00	0.000000E+00
2.600000	21.085840	0.000000E+00	0.000000E+00
2.700000	19.833370	0.000000E+00	0.000000E+00
2.799999	18.653680	0.000000E+00	0.000000E+00
2.899999	17.542420	0.000000E+00	0.000000E+00
2.999999	16.495570	0.000000E+00	0.000000E+00
3.099999	15.509360	0.000000E+00	0.000000E+00
3.199999	14.580240	0.000000E+00	0.000000E+00
3.299999	13.704900	0.000000E+00	0.000000E+00
3.399999	12.880240	0.000000E+00	0.000000E+00
3.499999	12.103330	0.000000E+00	0.000000E+00
3.599999	11.371440	0.000000E+00	0.000000E+00
3.699999	10.682000	0.000000E+00	0.000000E+00
3.799999	10.032570	0.000000E+00	0.000000E+00
3.899998	9.420874	0.000000E+00	0.000000E+00
3.999998	8.844763	0.000000E+00	0.000000E+00

			1of
4.099998	8.302208	0.000000E+00	0.000000E+00
4.199998	7.015791	0.000000E+00	2.003114E-01
4.299998	6.504248	1.587088E-01	1.942458E-01
4.399998	5.977942	2.753256E-01	1.878818E-01
4.499998	5.510567	3.540489E-01	1.824389E-01
4.599998	5.095648	4.006996E-01	1.775728E-01
4.699998	4.726803	4.209365E-01	1.730859E-01
4.799998	4.398181	4.198132E-01	1.688775E-01
4.899998	4.104669	4.015948E-01	1.649104E-01
4.999998	3.841931	3.697387E-01	1.611942E-01
5.099998	3.606340	3.269660E-01	1.577492E-01
5.199997	3.394952	2.753318E-01	1.546045E-01
5.299997	3.205383	2.163307E-01	1.517894E-01
5.399997	3.035687	1.510319E-01	1.493479E-01
5.499997	2.884278	8.015008E-02	1.473253E-01
5.599997	2.749908	4.106613E-03	1.457715E-01
5.699997	2.631436	-7.659547E-02	1.445610E-01
5.799997	2.526958	-1.603023E-01	1.433120E-01
5.899997	2.434266	-2.452841E-01	1.419336E-01
5.999997	2.351322	-3.299991E-01	1.403838E-01
6.099997	2.276308	-4.131188E-01	1.386381E-01
6.199996	2.207646	-4.935606E-01	1.366928E-01
6.299996	2.144015	-5.704918E-01	1.345587E-01
6.399996	2.084339	-6.433241E-01	1.322558E-01
6.499996	2.027776	-7.116890E-01	1.298102E-01
6.599996	1.973688	-7.754114E-01	1.272496E-01
6.699996	1.921612	-8.344782E-01	1.246014E-01
6.799996	1.871231	-8.889959E-01	1.218944E-01
6.899996	1.822338	-9.391606E-01	1.191508E-01
6.999996	1.774816	-9.852273E-01	1.163909E-01
7.099996	1.728600	-1.027484	1.136317E-01
7.199996	1.683668	-1.066240	1.108836E-01
7.299995	1.640030	-1.101803	1.081580E-01
7.399995	1.597709	-1.134467	1.054642E-01
7.499995	1.556729	-1.164514	1.028053E-01
7.599995	1.517125	-1.192201	1.001875E-01
7.699995	1.478921	-1.217760	9.761380E-02
7.799995	1.442143	-1.241400	9.508589E-02
7.899995	1.406808	-1.263309	9.260777E-02
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37.300000	1.485093	-1.599244	-1.410384E-02
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37.500000	1.484280	-1.598847	-1.361850E-02
37.600000	1.483871	-1.598649	-1.337187E-02
37.700000	1.483460	-1.598452	-1.312200E-02
37.800000	1.483046	-1.598257	-1.286903E-02
37.899990	1.482632	-1.598062	-1.261239E-02
37.999990	1.482213	-1.597868	-1.235214E-02
38.099990	1.481793	-1.597676	-1.208714E-02
38.199990	1.481370	-1.597484	-1.182016E-02
38.299990	1.480946	-1.597295	-1.154899E-02
38.399990	1.480520	-1.597107	-1.127353E-02
38.499980	1.480090	-1.596920	-1.099371E-02
38.599980	1.479658	-1.596734	-1.070929E-02
38.699980	1.479224	-1.596551	-1.042129E-02
38.799980	1.478786	-1.596369	-1.012780E-02
38.899980	1.478345	-1.596189	-9.829902E-03
38.999980	1.477903	-1.596011	-9.527512E-03
39.099980	1.477458	-1.595835	-9.219238E-03
39.199970	1.477008	-1.595661	-8.905144E-03
39.299970	1.476556	-1.595490	-8.585324E-03
39.399970	1.476100	-1.595321	-8.259334E-03
39.499970	1.475641	-1.595155	-7.926368E-03
39.599970	1.475178	-1.594992	-7.586495E-03
39.699970	1.474711	-1.594831	-7.239194E-03
39.799960	1.474239	-1.594674	-6.883518E-03
39.899960	1.473762	-1.594520	-6.519997E-03
39.999960	1.473281	-1.594369	-6.146389E-03
40.099960	1.472792	-1.594223	-5.763267E-03
40.199960	1.472299	-1.594080	-5.370494E-03
40.299960	1.471800	-1.593943	-4.965522E-03
40.399960	1.471293	-1.593810	-4.547326E-03
40.499950	1.470778	-1.593683	-4.115441E-03
40.599950	1.470255	-1.593562	-3.668632E-03
40.699950	1.469722	-1.593446	-3.204937E-03
40.799950	1.469178	-1.593339	-2.723122E-03
40.899950	1.468624	-1.593238	-2.219493E-03
40.999950	1.468055	-1.593146	-1.692263E-03
41.099950	1.467471	-1.593066	-1.138750E-03
41.199940	1.466868	-1.592995	-5.522370E-04
41.299940	1.466243	-1.592940	6.712226E-05
41.399940	1.465601	-1.592897	7.237979E-04
41.499940	1.464942	-1.592854	1.396837E-03
41.599940	1.464278	-1.592807	2.070386E-03
41.699940	1.463618	-1.592750	2.729499E-03
41.799930	1.462969	-1.592680	3.365042E-03
41.899930	1.462334	-1.592594	3.970317E-03
41.999930	1.461716	-1.592496	4.542008E-03
42.099930	1.461119	-1.592385	5.079654E-03
42.199930	1.460543	-1.592263	5.583453E-03
42.299930	1.459989	-1.592134	6.055246E-03
42.399930	1.459456	-1.591996	6.498110E-03
42.499920	1.458941	-1.591854	6.913925E-03
42.599920	1.458446	-1.591709	7.306622E-03
42.699920	1.457968	-1.591560	7.677304E-03
42.799920	1.457505	-1.591409	8.028515E-03
42.899920	1.457058	-1.591256	8.363643E-03
42.999920	1.456622	-1.591104	8.683452E-03
43.099910	1.456201	-1.590952	8.989996E-03
43.199910	1.455789	-1.590800	9.284188E-03
43.299910	1.455388	-1.590647	9.567777E-03
43.399910	1.454996	-1.590496	9.841777E-03
43.499910	1.454613	-1.590344	1.010676E-02

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43.599910	1.454237	-1.590193	1.036406E-02
43.699910	1.453869	-1.590042	1.061368E-02
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43.999900	1.452803	-1.589593	1.132236E-02
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44.199900	1.452121	-1.589296	1.176662E-02
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44.599890	1.450812	-1.588707	1.259945E-02
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44.799890	1.450184	-1.588416	1.299107E-02
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45.199880	1.448968	-1.587838	1.373451E-02
45.299880	1.448672	-1.587695	1.391179E-02
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45.599880	1.447802	-1.587267	1.442968E-02
45.699870	1.447518	-1.587125	1.459639E-02
45.799870	1.447237	-1.586985	1.476063E-02
45.899870	1.446959	-1.586844	1.492265E-02
45.999870	1.446684	-1.586704	1.508252E-02
46.099870	1.446410	-1.586564	1.524117E-02
46.199870	1.446137	-1.586425	1.539599E-02
46.299870	1.445869	-1.586286	1.554886E-02
46.399860	1.445603	-1.586148	1.570006E-02
46.499860	1.445339	-1.586010	1.584881E-02
46.599860	1.445078	-1.585873	1.599582E-02
46.699860	1.444818	-1.585736	1.614152E-02
46.799860	1.444560	-1.585599	1.628525E-02
46.899860	1.444304	-1.585464	1.642615E-02
46.999860	1.444052	-1.585329	1.656556E-02
47.099850	1.443801	-1.585194	1.670424E-02
47.199850	1.443553	-1.585060	1.684051E-02
47.299850	1.443304	-1.584926	1.697551E-02
47.399850	1.443060	-1.584793	1.710829E-02
47.499850	1.442817	-1.584661	1.724035E-02
47.599850	1.442576	-1.584529	1.736999E-02
47.699840	1.442336	-1.584397	1.749717E-02
47.799840	1.442100	-1.584267	1.762479E-02
47.899840	1.441864	-1.584136	1.775018E-02
47.999840	1.441630	-1.584006	1.787424E-02
48.099840	1.441399	-1.583877	1.799701E-02
48.199840	1.441168	-1.583748	1.811897E-02
48.299840	1.440938	-1.583619	1.823897E-02
48.399830	1.440711	-1.583491	1.835712E-02
48.499830	1.440487	-1.583364	1.847461E-02
48.599830	1.440263	-1.583237	1.859141E-02
48.699830	1.440040	-1.583110	1.870685E-02
48.799830	1.439819	-1.582984	1.882121E-02
48.899830	1.439599	-1.582859	1.893365E-02
48.999820	1.439381	-1.582735	1.904452E-02
49.099820	1.439167	-1.582610	1.915560E-02
49.199820	1.438951	-1.582486	1.926526E-02
49.299820	1.438739	-1.582363	1.937333E-02
49.399820	1.438526	-1.582240	1.948095E-02
49.499820	1.438316	-1.582119	1.958690E-02
49.599820	1.438107	-1.581997	1.969198E-02
49.699810	1.437900	-1.581875	1.979690E-02
49.799810	1.437693	-1.581755	1.989900E-02