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Designated Original Redacted Version Public Available

July 10, 2007

Mr. Alexander Adams
U. S. Nuclear Regulatory Commission
Research and Test Reactors Branch A
Office of Nuclear Reactor Regulation
Mail Stop O12-G13
One White Flint North
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Reference: Oregon State University TRIGA Reactor (OSTR)
Docket No. 50-243, License No. R-106
Request for Additional Information (RAI) Regarding License Renewal, Oregon State University TRIGA Reactor (TAC NO. MC5155) dated May 21, 2007

Subject: Oregon State University Response to RAI Regarding License Renewal, Oregon State University TRIGA Reactor dated May 21, 2007

Mr. Adams:

In a letter dated May 21, 2007, the U.S. Nuclear Regulatory Commission (NRC) requested that Oregon State University (OSU) provide additional information with regards to the OSU license renewal application of October 5, 2004, as supplemented. Enclosed is the OSU response to the request. If you have any questions, please call me at the number above. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 7/10/07.

Sincerely,

Steve Reese
Director

Enclosure

cc: Document Control, NRC
Al Adams, NRC
Craig Bassett, NRC
John Cassady, OSU
Rich Holdren, OSU
Todd Palmer, OSU
Mike Hartman, OSU

A020

Oregon State University

Responses to RAI Letter of May 21, 2007

1. Section 2.2.1. Figure 2.2 shows a rail line in the vicinity of the reactor facility. Explain why shipments on this rail line do not pose a threat to the reactor facility.

   The rail line is located approximately 1200 feet to the south of the Radiation Center. The reactor building is located on the north side of the Radiation Center. Based on the distance and the location of the reactor building on the opposite side of the Radiation Center from the rail line, shipments on the rail line do not pose a threat to the reactor facility. With the exception of a single rail car a year, all transport on the rail spur involves lumber.

2. Section 2.3. What would be the source of site meteorology data in case of an emergency situation?

   The reactor building has a wind speed and direction anemometer. The data for the anemometer is collected by a data logger and recorded on a computer hard drive. The data is time stamped so for any release, the wind speed and direction data could be downloaded, averaged over a given time period, and entered in the Gaussian plume model calculations used in Chapter 13 of the SAR.

3. Section 3.4. What is the relationship between the UBC 1964 Zone 3 seismic requirements and the maximum ground accelerations given in Table 2-4?

   This question will be answered at a later time.

4. Section 4.2.1.9. The discussion of fuel swelling at high burnups refers to the agglomeration of fission gases at room temperatures above 1300°F. This swelling is time and temperature dependent. Provide a discussion if there should be a steady state temperature limit to control this type of swelling.

   This question will be answered at a later time.

5. Section 4.3. What is the minimum reactor coolant leakage that can be detected from the reactor pool? What is the largest amount of primary coolant that can be lost from the reactor pool without detection? What is the probable leakage path? Discuss the impact that any potential leakage might have on the health and safety of the public or the environment.

   A loss of approximately 10 gallons a week would be the minimum detectable leakage rate. This is based upon historical primary water makeup rates that typically run 25 gallons per week for normal operations. These numbers are approximate because they are heavily dependent upon such things as operational history, temperature and humidity.

The largest amount of primary water that could be lost from the reactor pool without detection is approximately 40 gallons. The water level is routinely kept at 2 inches above the low water level set point. One inch is equivalent to approximately 20 gallons. The water level monitor is continuously in operation and is monitored by the reactor operator during the day and remotely during non-working hours.

It is assumed that this question deals with an unknown failure of the primary water tank integrity such as a failed weld joint or a pin-hole defect. The probable leakage path would run from the primary tank, to the gap between the tank liner and the bioshield, to a beam port or the thermal column and out onto the reactor bay floor. From here the water would be collected and analyzed. However, as describe in Chapter 11, typical radioactivity concentrations in the primary water are very low. In fact, they are nominally equal to or lower than the limits for water effluent concentration found in Column 2, Table 2, Appendix B of 10 CFR 20. To reiterate, the source term in this case is lower than that allowed for environmental release as described in 10 CRF 20 and therefore could not negatively impact the health and safety of the public or the environment.

6. What is the maximum fuel element power for possible core loadings (#8)? What are the peaking factors? Tables 4-11 and 4-12 only contain average power per element.

   This question will be answered at a later time.

7. Section 5.3. Is secondary pressure higher than primary pressure when both pumps are shut down? If not, what is the impact of a primary to secondary heat exchanger leak?

   The secondary pressure is higher than the primary pressure when both pumps are shut down. The pressure on the primary side is estimated at 8.6 psi, conservatively assuming a maximum of 20 feet of head. The pressure on the secondary side is estimated at 13.0, again conservatively assuming a minimum of 30 feet of head. The differential pressure is then 4.4 psid. The actually differential pressure is likely double this number using more realistic elevations.

8. Section 5.5. Is there any physical connection between the city water system and the reactor primary system? If so, how is the possibility of siphoning primary water into the city water eliminated?

There is no direct physical connection between the city water system and the reactor primary water system. Water is added to the primary only after it has passed through a reverse osmosis (RO) unit designed to purify the water. Water is passed from one side of the RO unit to the other by dripping into a flask, thereby precluding the possibility of siphoning. Additionally, all water entering the building is protected against siphoning with a reverse pressure backflow prevention device which is inspected annually.

9. Section 7.7.3. Is the primary water activity monitor always on line when the reactor is in operation? If not, how would fuel fission product leakage be detected by the reactor operator?

The primary water activity monitor is normally continuously in operation when the reactor is in operation. However, the monitor is not very sensitive and it is not intended to be our primary means of detecting a fuel leak. Any fuel fission product leakage would likely be detected (i.e., reaching the alert and alarm levels on the instruments and alerting the reactor operator) on the reactor top continuous air monitor first, the stack continuous air monitor second and the area radiation monitor on the demineralizer tank third, in that order, before the primary water activity monitor.

In addition to the primary water activity monitor, the water is sampled monthly for activity by analysis with gamma spectroscopy, dried on a filter and analyzed on a gas flow proportional counter for beta activity, and analyzed with a liquid scintillation counter for low energy beta activity.

10. Section 9.5. Describe what parts of the Radiation Center are under the jurisdiction of the reactor license. For example, Section 9.1.2 discusses laboratories. However, TS 5.1.a discusses the restricted area.

All activities inside the TS 5.1.a defined restricted area fall under the jurisdiction of the reactor license. All of the laboratories referred to in Section 9.1.2 are in the reactor building and are under the jurisdiction of the reactor license.

11. Chapter 9. Verify that the current material possession limits in the license are to be carried over into the renewed license unchanged. Discuss possession and storage of the AGN core. For example, how is sub-criticality of the AGN core assured? Is the AGN core subject to TS 5.4?

The current material possession limits will be carried over into the renewed license unchanged.

The core of the AGN was placed [redacted] to secure it in place. The control rods were removed and shipped to another licensee. Rolled up cadmium sheet was placed in the glory hole and the vacant control rod positions. The [redacted] core is secured to prevent removal [redacted] The AGN core is subject to TS 5.4. Data from the Idaho State University AGN-201 reactor shows that the k-effective for the reactor without the control rods is 0.967 (Personal communication with John Binnion, Reactor Administrator, ISU AGN-201 Research Reactor). We are confident that without the control rods and the insertion of the cadmium, the k-effective is less than 0.9.

12. Chapter 9. Describe the facility compressed air system.

Filtered and regulated compressed air is provided to the reactor building for general use and to operate the transient rod air cylinder. The primary system utilizes a 7.5 Hp 480 VAC 3 phase skid mounted two-cylinder two-stage reciprocating compressor, water separator/cooler and a pressure tank located in room D104A. An outlet pressure regulator maintains the supply air pressure at 80 psig. Low pressure alarms annunciate on the reactor control room annunciator panel to provide indication of inadequate air supply. Backup service can be provided, if needed, by cross connecting the distribution system to the two compressor systems used for the Radiation Center.

13. Section 4.5.3.1.3 reaches a conclusion of a safety limit of 2,012°F (1,100°C) for cladding temperature less than 930 F (500°C). However, your proposed Technical Specifications contain different safety limits. Please clarify.

This is a typographical error. This safety limit should be 2,100°F (1,150°C).

14. Section 4.6. Provide additional information on neutronic and thermal-hydraulic analysis that demonstrates that sufficient safety margins exist during operation at your licensed power level.

This question will be answered at a later time.

15. Section 7.3.3. Briefly describe the basis for interlocks.

For the transient rod interlock, section 13.2.2.2.1 shows that the designed limiting reactivity insertion for [redacted] fuel is \$2.59 at the end of core life. This interlock will limit transient rod reactivity insertions below this value. Furthermore, this interlock is designed such that if the electrical (i.e., limit switch) portion fails, a mechanical (i.e., metal bracket) will still keep the reactivity insertion below the criterion.

The 1-kW permissive interlock to prevent pulsing when wide range log power is above 1-kW is unnecessary. Section 13.2.2.2.1 shows that the peak temperature reached during an end-of-life [redacted] core was 1,150°C for an initial fuel temperature of 20°C. The methodology clearly shows that if the initial temperature was higher, the resulting peak temperature must be lower. Therefore, the 1-kW permissive interlock is not performing a safety function and is not needed in the TS. We therefore would like to remove it.

In pulse mode, it is necessary to limit the reactivity inserted to less than the design limit of $2.59 at the end of core life analyzed in 13.2.2.2.1. This interlock ensures that all pulse reactivity is due to only the transient rod while in pulse mode. Otherwise, any control rod removal in pulse mode would add to the inserted reactivity of the transient rod and create an opportunity for exceeding the reactivity insertion limit.

The single rod withdrawal interlock prevents a situation where by the ramped (non-pulse) reactivity insertion rate could exceed that analyzed in section 13.2.2.2.2. That analysis shows that the reactivity insertion from the removal rate of the most reactive rod is still well below the reactivity insertion design limit of $2.59. However, this analysis only looks at removal of a single control rod, this case the most reactive control rod. It does not take into account multiple control rods being removed simultaneously.

The rod withdrawal prohibit interlock prevents the operator from adding reactivity in the following situations:

a) When the count rate on the wide-range log power channel falls below 2 cps, the count rate is insufficient to produce meaningful instrumentation response. If the operator were to insert reactivity under this condition, the period could rapidly become very short and result in an inadvertent power excursion. A neutron source is added to the core to create sufficient instrument response that the operator can recognize and respond to changing conditions.
b) When the period/log test switch is out of the operate position, a false signal is fed into the signal chain for the wide-range log and wide-range linear channels, effectively rendering those technical specification required measuring channels inoperable.
c) When the detector current selector switch is out of the operate position, the signal for the selected detector is diverted to an ammeter, effectively rendering the selected technical specification required measuring channels inoperable.
d) When the fuel temperature selector switch is in the fourth position and not one of the three positions to read the thermal couples in the instrumented fuel element, the technical specification required measuring channel is effectively rendered inoperable.

16. Section 7.4.1. Briefly describe the values and basis for scram circuit set points.

The manual scram must be functional at all times the reactor is in operation. It has no specified value for a scram set point. It is initiated by the reactor operator manually.

The fuel element temperature scram causes a scram in excess of the LSSS, which is 510°C. The supporting arguments for the safely limit of 1150°C for FLIP fuel are given in section 4.5.3.1. The LSSS is set to less than half for the safety limit. This is more than adequate to account for uncertainties in instrument response and core position of the instrumented fuel element.

The set point for both the safety and percent power channels are normally set to 106% of 1 MW(t), which is below the licensed power of 1.1 MW(t). The 6% difference allows for expected and observed instrument fluctuations at the normal full operating power of 1 MW(t) to occur without scramming the reactor unnecessarily. Conversely, section 13.2.2.2.2 shows that this set point is more than sufficient to prevent exceeding the reactivity insertion limit during non-pulsing operations and prevent the operator from inadvertently exceeding the licensed power.

The reference to the period channel scram needs to be deleted.

The high voltage scram must be set to initiate a scram when the high voltage for any of the three detectors drops to 25% or less of the nominal operating voltage. The loss of operating voltage down to this level is an indication of detector failure. Many measuring channels and safety systems are fundamentally based upon accurate response of the detectors.

The external scram set points vary depending on the experiment involved. Typically, this scram is used for experiments were automatic shutdown of the reactor is necessary to reduce the intensity of a radiation field.

17. Section 10.2.2. The rotating rack represents a different type of moving experiment where there can be continuous reactivity changes while the rack is in motion as opposed to a moveable experiment where it is assumed that the movement is into and out of the reactor. Discuss if a TS limit is needed (reactivity change per unit time) on moving experiments.

Experiments placed in the rotating rack will be considered secured experiments because the experiments will not be moving into or out of the core. Additionally, extensive experience has shown that experiments in the rotating rack have little (i.e., ~$0.10 at most) or no measureable reactivity worth regardless of the material composition.

Because it has been shown in section 13.2.2.2.1 that the reactor is protected when limited to pulses of up to $2.55, shown in section 13.2.2.2.2 that the existing safety channels adequately protect the reactor from ramped reactivity insertions, and movable experiments are limited to $1.00, any TS limit on moving experiments would be redundant.

18. Section 10.3. Provide a basis for experimental TS limits.

The reactivity limit of $1.00 for movable experiments is designed to prevent an inadvertent pulse from occurring. Movable experiments are by their very nature experiments in a position were it is possible for a sample to be inserted or removed from the core. Not withstanding any other requirements, such as shutdown margin, it is prudent to limit the reactivity worth of these experiments below which a pulse could occur. That being said, Section 13.2.2.2.1 clearly shows that this value is still below the analyzed design limit of $2.59 for end of life [redacted] fuel.

The reactivity worth limit of $2.55 for any single experiment is designed to prevent an inadvertent pulse from exceeding the design limit of $2.59 for end of life [redacted] fuel. This limit applies to both movable and secured experiments. Regardless of any other administrative or physical requirements, this limit has been shown in Section 13.2.2.2.1 to protect the reactor during the fuel's entire lifetime.

The reactivity worth limit of $3.00 for the total worth of all experiments is designed to ensure shutdown of the reactor upon removal of all experiments. The technical specifications require that the OSTR shutdown margin be at least $0.55 with the most reactive rod withdrawn. Conversely, this also represents the minimum worth of the remaining control rods. The transient rod is the most reactive rod, with a worth of approximately $4.00. Therefore with all the rods inserted, the reactor is shutdown by at least $4.55. If all experiments were removed at the same time, the reactor would still be shutdown by at least $1.55. The transient rod is not considered removed from the core and its worth is included in this calculation. The implied assumption with the $3.00 limit is that you are adding negative reactivity with the insertion of the experiments into the core. This limit is designed to ensure that the reactor can remain shutdown upon the removal of the experiments. The reactor would be required to shutdown because moveable experiments are limited to less than $1.00, with the remaining being secured experiments. By definition, the reactor must be shutdown to remove the secured experiments. At the same time, this limit does not absolve responsibility for meeting the shutdown requirement during operation.

The basis for the 25 mg limit on explosive materials for experiments is described in Section 13.2.6.2. This analysis shows that the limit is safe provided that the proper container material with appropriate diameter and wall thickness are used.

We would like to remove the TS 3.8.2.b. Additionally, we would like to remove the sixth paragraph of section 13.2.6.2.

For failures and malfunctions of experiments, the limit is to that which will not result in exceeding the applicable dose limits in 10 CFR 20. This is regulation and needs no further justification. The assumptions used in TS 3.8.3 are all endorsed in NUREG-1537 as, "Such specifications (assumptions) ensure conservatism in the safety analysis of the experiment."

19. Table 11-4. Provide the alarm basis for TS required radiation monitors.

Continuous Air Monitor-Reactor Top-Airborne Particulate: A fraction (0.06%) of the DAC for Cs-138 is used for the alarm set-point. This value permits early detection of a cladding failure and yet is high enough to prevent spurious alarms due to fluctuations in the background count rate.

Continuous Air Monitor-Effluent Stack-Airborne Particulate: The alarm set-point for the particulate channel is normally set to the net count rate equivalent of 6.7×10^{-8} $\mu Ci\ cm^{-3}$ for Cs-138 which is 0.3% of the DAC and 83% of the annual effluent concentration limit (Table 2, App. B, 10 CFR 20). This value permits early detection of a cladding failure and yet is high enough to prevent spurious alarms due to fluctuations in the background count rate.

Continuous Air Monitor-Effluent Stack-Gas: The alarm set-point for the gas channel is normally set to the net count rate equivalent of 4×10^{-6} $\mu Ci\ cm^{-3}$, which is the technical specifications annual average concentration limit for ^{41}Ar. Section 11.1.1.1.1 shows that an abnormally high release rate (i.e., 11 $\mu Ci\ s^{-1}$ which corresponds to 6.3×10^{-7} $\mu Ci\ cm^{-3}$) meets the requirements of NESHAP and 10 CFR 20. This value permits early detection of a cladding failure and yet is high enough to prevent spurious alarms due to fluctuations in the background count rate.

Area Radiation Monitors: The alarm basis for the area radiation monitors are set at levels agreed upon by the Reactor Supervisor and the Senior Health Physicist based on anticipated normal or abnormal radiation levels.

20. Section 13.2.2.2.2. Does this analysis consider the 2-second TS limit on scram time? If not, please discuss. Also, Section 4.2.2 contains information on reactivity insertion rates that is different from that given in Table 13-12. Please explain.

The analysis in section 13.2.2.2.2 does not consider the 2-second TS limit on scram time. Our interpretation of this TS limit is that it starts upon the signal initiation and ends when the slowest rod is in its fully down position, including the signal processing time of the instrument channel. The 0.5 seconds in the section is approximately equal to instrument response time that is the slowest, in this case it is the fuel element temperature channel. However, use of 0.5-seconds is appropriate because control rods begin to insert at this point and power immediately begins to turn.

21. Section 13.2.3.2.2. Discuss your policy on the handling of heavy loads in the reactor room.

The lifting of heavy loads in the reactor bay is covered under Oregon State TRIGA Reactor Operating Procedure 23, *Crane Operation Procedures*. This procedure details the authorization for use of the crane, rigging and lifting procedures, and limitations of use. That procedure will include language stating that the movement of heavy loads over the bioshield will be prohibited except as part of a fulfillment of a task that would specifically require it and the movement is approved by the Reactor Supervisor.

22. Section 13.2.6.2. This section contains a discussion of the $3.00 reactivity limit worth of experiments. Your proposed TS contains a minimum shutdown margin of $0.55 while this section of the SAR has a value of $0.57. Please clarify. The statement is made that with all the rods inserted, the reactor is shut down by $4.62. Does this value have the transient rod fully withdrawn from the core? If not, show how the shutdown margin is met with the transient rod (or the rod of highest worth if not the transient rod) fully withdrawn.

The value in section 13.2.6.2 should be $0.55 not $0.57. Additionally, we recommend changing the fifth paragraph to read:

> "A further limit on the reactivity worth of all experiments has been set at $3.00. The technical specifications require that the OSTR shutdown margin be at least $0.55 with the most reactive rod withdrawn. Conversely, this also represents the minimum worth of the remaining control rods. The transient rod is the most reactive rod, with a worth of approximately $4.00. Therefore with all the rods inserted, the reactor is shutdown by at least $4.55. If all experiments were removed at the same time, the reactor would still be shutdown by at least $1.55. The transient rod is not considered removed from the core and its worth is included in this calculation. The implied assumption with the $3.00 limit is that you are adding negative reactivity with the insertion of the experiments into the core. This limit is designed to ensure that the reactor can remain shutdown upon the removal of the experiments. The reactor would be required to shutdown because moveable experiments are limited to less than $1.00, with the remaining being secured experiments. By definition, the reactor must be shutdown to remove the secured experiments. At the same time, this limit does not absolve responsibility for meeting the shutdown requirement during operation."

23. Section 13.2.7.2. This section discusses the use of emergency power at the facility. Is emergency power needed to ensure public health and safety? If not, please explain. If so, discuss the need for emergency power TSs.

The emergency electrical power system is not necessary to safely shutdown the reactor. It is not needed to ensure public health and safety. In the event of a loss of electrical power without an emergency power, all rods would insert into the core automatically due to a loss of power to the electromagnets, the ventilation would shutdown, and although the primary and secondary water pumps would stop, the amount of primary water is more than sufficient to dissipate the decay heat. Additionally, we can easily verify shutdown of the reactor manually by visually inspecting the core from the reactor top. With this in mind, the last paragraph of Section 13.2.7.2 should be deleted to make it consistent with Chapter 8 and true to the intent.

24. TS 2.2. Discuss the derivation of the limited safety system setting (LSSS) value of 510°C. Discuss how LSSS protects fuel from exceeding the safety limit considering issues such as instrumented fuel element placement in the core versus the core hot spot, the thermocouple placement in the instrumented fuel element versus the fuel element hot spot, the accuracy of the measuring instrumentation and transient behavior of the reactor safety system.

This question will be answered at a later time.

25. TS 3.1.3. Discuss the amount of excess reactivity needed for continuing reactor operation. Include values for such uses of excess reactivity as power defect, experiments and burnup.

TS 3.1.3 should be revised to be limited to $9.00. During routine operations we observed a power defect of approximately $2.50, irradiation facilities worth approximately $2.50 (i.e., CLICIT) and fission poison (i.e., xenon and samarium) build-up of approximately $1.00. If the total value for all experiments of $3.00 is added in, the total excess reactivity needed equals $9.00. This is deemed sufficient to continue reactor operations for all foreseeable operation needs.

26. TS 3.2.2. It appears that the requirement for a period-circuit and safety power level measuring channels in the current TSs are removed from your proposed TSs. Please provide a justification for the removal of these measuring channels.

The period circuit is not required because the reactor is specifically designed to be safely pulsed up to $2.55, as shown in section 13.2.2.2.1. Additionally, the consequences of an uncontrolled withdrawal of a control rod has been analyzed in section 13.2.2.2.2 and was shown to be far less than the $2.55

reactivity insertion limit. As it does not serve a safety function, it should not be a TS required measuring channel.

We agree that the safety power level measuring channel should be included in Table 1 as before.

27. TS 3.2.3., Table 2. Please justify the changes made between Table 1 of your current TSs and Table 2 of your proposed TSs.

We agree that both the safety and percent power level measuring channels should be included in Table 2 with a requirement for a scram setpoint at or below 1.1 MW(t) in steady state and square wave mode.

The fuel element temperature safety channel is not needed in pulse mode because it does not serve a protective function. Once a pulse is initiated, the only thing limiting the pulse height, total power, and resulting fuel temperature is the \$2.55 maximum limit on reactivity insertion. The fuel element temperature safety channel inevitably follows the pulse (i.e., in terms of pulsing, it is after-the-fact) and is not capable of limiting it.

We agree that the preset timer with a requirement to perform a scram on the transient rod 15 seconds or less after a pulse in pulse mode should be included in Table 2.

28. TS 3.7.1. Your proposed TS allowed the reactor to be operated with certain radiation monitors out-of-service if certain conditions are met. Please provide a basis.

We propose to change TS 3.7.1.a to read, "The exhaust gas and exhaust particulate radiation monitors are operating." The loss for short periods of time of the continuous air particulate monitor and/or area radiation monitors are adequately covered by the dual particulate and gas detection capability on the exhaust gas system as all air in the reactor bay must pass by its sampling location. Two hours was felt to be a reasonable amount of time to perform repairs or maintenance. [redacted]

29. TS 3.8.2.a. Provide a basis for the irradiation of 0.014 lbs-equivalent of the TNT in the laboratory area.

The basis for the irradiation of 0.014 lbs-equivalent of TNT in the laboratory area was discussed and approved in amendment #3 for license R-106. The facility and activities described in the amendment are no longer applicable or relevant. We request that the exception for the irradiation of the TNT found in TS 3.8.2.a be removed.

30. Section 11.1.1.1.1. Is it possible that some one near the fence line could receive exposure from the shine from release cloud passing overhead? If so, how does this dose compare against the dose from cloud immersion?

Using the highest release rate of 11 μCi s^{-1} and a volumetric flow rate of 4.4 x 10^{-6} cm^3 s^{-1}, the concentration would be 2.5 x 10^{-6} μCi cm^{-3}. The fence line is 17 m from the reactor building and the highest diffusion parameter would be 1.36 m. The effective stack height is 32 m. Using Microshield 5.05, a cylinder with a diameter of 1.36 m, length of 34 m and concentration of 2.5 x 10^{-6} μCi cm^{-3} is used as the source and the dose point is located 17 m along the length of the cylinder and 32 m from the cylinder. The dose rate at this location is 6.88 x 10^{-5} mR hr^{-1}. The length of the cylinder is twice the distance from the wall to the fence line to take into account shine from the passing plume. A sensitivity analysis showed no significant increase past this point. The highest diffusion parameter at 34 m is 2.72 m. Changing only the diameter of the source cylinder, results in a dose rate of 2.75 x 10^{-4} mR hr^{-1}.

The same release rate and stack height were used in COMPLY 1.5d, which takes into account immersion, and resulting dose rate was 0.5 mrem yr^{-1} or 5.71 x 10-5 mrem hr^{-1}.

31. Sections 11.1.3 and 11.1.5.6. Dosimetry issue guidelines are similar to ALARA investigative limits. Should dosimetry issue guidelines be at a lower dose than the ALARA investigative limit? If not, please explain.

Section 11.1.3, bullet two, should read exposure investigations being initiated when an individual receives a dose in any reporting period greater than 1% of the applicable regulatory limit for those individuals who received dosimetry.

32. Section 11.2.3. Describe the liquid waste system. Describe operational or design features to ensure that non soluble radioactive material is not released into the environment.

All non-sewer drains within the reactor building drain to a 3000 gal hold-up tank. The tank has a level indicator with high and full alarms which are monitored by reactor operators during the day and remotely at night. The hold-up tank is sampled and tested for radioactive materials on a monthly basis. The samples are counted on a HPGe detector for gamma analysis, filtered and distilled samples for liquid scintillation counting primarily for tritium, and filter and filtrate samples on a windowless gas flow proportional counter. If any of the analyses are above the lower limit of detection for the appropriate detector, a solubility determination is performed. Initially, two planchets are weighed. One hundred milliliters in small increments are added directly to one planchet and slowly evaporated at just below boiling temperature. Drying is completed in an oven and cooled in a desiccator. The sample is weighed just before counting. At least 100 mL are filtered through a series of filters with the last being of 0.45 μm. The filtrate is then processed through the same evaporation and drying process as above. The prepared samples are then counted on a windowless gas flow proportional counter to determine the gross alpha and beta count rates. The lower limit of detection at the 95% confidence level is calculated for the difference between the filtered and unfiltered samples. If the difference is below the lower limit of detection for both alpha and beta, the holdup tank water contains no insoluble radioactive material and therefore meets the solubility criteria for discharge to the sewer.

33. Section 13.2.1.1. For the Maximum Hypothetical Accident Scenarios B and C, is exposure from building shine from the source term inside the reactor room considered? If not, discuss why this is not a significant contributor to dose outside the reactor room.

Microshield version 5.05 was used to determine the dose rates from each isotope in Table 13.1 with and without pool water. The dose rate at each distance for each isotope was used in the following equation to determine the total dose:

$$\text{Dose} = \sum_i \frac{\text{Dose rate}_i \times \left(1 - e^{-\lambda_{eff} t_{st}}\right)}{\lambda_{eff}}$$

t_{st} = stay time of personnel and $\lambda_{eff} = \lambda_i + \lambda_v$ where λ_i is the decay constant for the ith isotope and λ_v is the ventilation constant for the appropriate scenario. The initial dose is at t = 0 and the other is for the exposure time for each scenario. The thyroid dose was calculated by multiplying the total by 0.03.

RX Bay Volume Source Scenario A (Microshield)

Distance from Wall Location (m)	Pool Water (mR)			
	Thyroid DDE	Whole Body	Thyroid DDE (after 8.52 s)	Whole Body (after 8.52 s)
10	8.97E-06	2.99E-04	1.16E-07	3.87E-06
50	1.06E-06	3.54E-05	1.37E-08	4.58E-07
100	2.54E-07	8.48E-06	3.30E-09	1.10E-07
150	9.59E-08	3.20E-06	1.25E-09	4.16E-08
200	4.44E-08	1.48E-06	5.79E-10	1.93E-08
250	2.31E-08	7.70E-07	3.02E-10	1.01E-08

RX Bay Volume Source Scenario A (Microshield)

Distance from Wall Location (m)	W/O Pool Water (mR)			
	Thyroid DDE	Whole Body	Thyroid DDE (after 8.52 s)	Whole Body (after 8.52 s)

10	1.35E-05	4.49E-04	2.10E-07	7.00E-06
50	1.60E-06	5.32E-05	2.48E-08	8.27E-07
100	3.79E-07	1.26E-05	6.01E-09	2.00E-07
150	1.42E-07	4.74E-06	2.29E-09	7.64E-08
200	6.50E-08	2.17E-06	1.07E-09	3.57E-08
250	3.35E-08	1.12E-06	5.62E-10	1.87E-08

RX Bay Volume Source Scenario B (Microshield)

Distance from Wall Location (m)	Pool Water (mR)			
	Thyroid DDE	Whole Body	Thyroid DDE (after 14.7 m)	Whole Body (after 14.7 m)
10	5.55E-04	1.85E-02	1.78E-04	5.93E-03
50	6.57E-05	2.19E-03	2.10E-05	7.02E-04
100	1.57E-05	5.23E-04	5.05E-06	1.68E-04
150	5.91E-06	1.97E-04	1.91E-06	6.36E-05
200	2.73E-06	9.12E-05	8.83E-07	2.94E-05
250	1.42E-06	4.74E-05	4.60E-07	1.53E-05

RX Bay Volume Source Scenario B (Microshield)

Distance from Wall Location (m)	W/O Pool Water (mR)			
	Thyroid DDE	Whole Body	Thyroid DDE (after 14.7 m)	Whole Body (after 14.7 m)
10	8.70E-04	2.90E-02	2.25E-04	7.51E-03
50	1.03E-04	3.44E-03	2.67E-05	8.89E-04
100	2.43E-05	8.11E-04	6.38E-06	2.13E-04
150	9.05E-06	3.02E-04	2.40E-06	7.99E-05
200	4.10E-06	1.37E-04	1.11E-06	3.68E-05
250	2.10E-06	7.00E-05	5.73E-07	1.91E-05

RX Bay Volume Source Scenario C (Microshield)

Distance from Wall Location (m)	Pool Water (mR)			
	Thyroid DDE	Whole Body	Thyroid DDE (after 14.7 m)	Whole Body (after 14.7 m)
10	6.17E-03	2.06E-01	6.07E-03	2.02E-01
50	7.33E-04	2.44E-02	7.21E-04	2.40E-02
100	1.70E-04	5.67E-03	1.68E-04	5.59E-03
150	6.22E-05	2.07E-03	6.15E-05	2.05E-03
200	2.81E-05	9.37E-04	2.78E-05	9.28E-04
250	1.43E-05	4.78E-04	1.42E-05	4.74E-04

RX Bay Volume Source Scenario C (Microshield)

Distance from Wall	W/O Pool Water (mR)

Location (m)	Thyroid DDE	Whole Body	Thyroid DDE (after 14.7 m)	Whole Body (after 14.7 m)
10	1.39E-02	4.62E-01	1.26E-02	4.19E-01
50	1.65E-03	5.50E-02	1.49E-03	4.98E-02
100	3.77E-04	1.26E-02	3.44E-04	1.15E-02
150	1.36E-04	4.53E-03	1.25E-04	4.18E-03
200	5.94E-05	1.98E-03	5.53E-05	1.84E-03
250	2.95E-05	9.83E-04	2.77E-05	9.22E-04

34. Section 13.2.1.1. A reactor room leak rate of 1.69x 10^4 $cm^3 sec^{-1}$ is used based on the original August 1968 SAR. The original SAR does not contain a basis for the leak rate. Please discuss the leak rate.

The leakage from the room is through the walls brought about by a pressure differential between the room and outside. This pressure differential was assumed to arise through the unlikely combination of a drop in atmospheric pressure of 1.5" Hg and an increase in room temperature of 40°C in 12 hours. Using the ideal gas law, $P_1V_1/T_1= P_2V_2/T_2$, where temperature is in K, pressure is in inches of Hg, and volume is in cubic centimeters. The initial conditions of T_1=295.2 K (22.2 °C) and P_1=29.4" Hg in the RX bay of volume 3.74E4 cm^3, the increase in temperature of 40°C would result in the pressure increasing to 33.4" Hg. The atmospheric drop 0f 1.5" Hg would mean an increase in room pressure to 34.9" Hg and assuming an outside temperature of 8.7°C, the resulting bay volume would be 3.01E9 cm^3. The leak rate is determined by the difference in the initial and final bay volumes which is 7.3E8 cm^3 divided by 12 hrs. or 43200 sec to equal 1.69E4 cm^3 sec^{-1}.

35. Table 11-3. The ninth item on this table is under what license?

That item is on the State of Oregon license, ORE90005.

36. Section 15.2. Please update your financial information to include FY 2007 to FY 2012. Clearly show projected revenue sources and expenses for each year such that the revenue sources cover expenses.

Here is an update of Tables 15-1 and 2 to reflect projected sources and expenses out to FY 2012.

Table 15-1 Summary of Expenses

	FY 08	FY 09	FY 10	FY 11	FY 12
Unclassified Salary	$ 232,802	244,442	256,664	269,498	282,973
Classified Salary	$ 147,050	154,403	162,123	170,229	178,741
Student Wages	$ 6,330	6,646	6,979	7,327	7,694
Payroll Expenses	$ 152,015	159,616	167,597	175,976	184,775

Services & Supplies	$	70,822	74,363	78,081	81,985	86,084
Travel	$	277	290	305	320	336
Equipment	$	11,487	12,061	12,664	13,297	13,962
Total	$	620,782	651,821	684,412	718,633	754,565

Table 15-2 Summary of Income

	FY 08	FY 09	FY 10	FY 11	FY 12
State E&G Funds	$ (763,161)	$ (801,319)	$ (841,385)	$ (883,454)	$ (927,627)
Returned Overhead	$ (30,000)	$ (31,500)	$ (33,075)	$ (34,729)	$ (36,465)
Salary Redistributions	$ (20,000)	$ (21,000)	$ (22,050)	$ (23,153)	$ (24,310)
Grants and Awards	$ (85,000)	$ (89,250)	$ (93,713)	$ (98,398)	$ (103,318)
	$ (898,161)	$ (943,069)	$ (990,223)	$(1,039,734)	$(1,091,720)

37. Section 15.3. The NRC is treating license renewal as the issuance of a new license. As such, the University must submit new decommissioning financial assurance. The regulations in 10CFR 50.75(e)(iv) permits licensees to provide assurance of decommissioning funding by "a statement of intent containing a cost estimate for decommissioning, and indicating that funds for decommissioning will be obtained when necessary." The staff notes that the University has used a statement of intent for the decommissioning funding assurance for their current license. The statement of intent must be signed by an official who has the authority to commit to spending the necessary funds to accomplish decommissioning, and it should clearly asserted in the statement of intent that the signing official has that authority. In addition, the statement of intent should contain a statement that funding will be provided sufficiently in advance of decommissioning to prevent delay of required activities. If decommissioning funding is to continue to be assured by Oregon State University, submit an updated statement of intent to this effect, signed by an appropriate State official. Otherwise, 10 CFR 50.75(e) provides alternate options for assurance of decommissioning funding.

    A statement of intent will be submitted separately.

38. Section 10.3. Term "unreviewed safety question" is not used anymore. Please update using current terminology.

    The sentence referred to, "The ROC review and approval process also follows the provisions of 10 CFR 50.59 to ensure that the proposed experiment does not constitute an unreviewed safety question and does not require a change in the Technical Specifications.", should be replaced with, "The ROC review and approval process also follows the provisions of 10 CFR 50.59 and the guidance found in Regulatory Guide 1.186."

39. Section 10.1. Reference is made to figures in Chapter 1. Chapter 1 contains no figures. Please clarify.

The second sentence in Section 10.1 should be deleted. The referenced figures were removed prior to submission and will not be replaced.