OFFICE OF NUCLEAR REACTOR REGULATION

REGULATORY AUDIT QUESTIONS

REGARDING RENEWAL OF FACILITY OPERATING LICENSE NO. R-130

FOR THE UNIVERSITY OF CALIFORNIA DAVIS

MCCLELLAN NUCLEAR RESEARCH CENTER TRIGA REACTOR

DOCKET NO. 50-607

By letter dated June 11, 2018, the Regents of the University of California (licensee) submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC) for a 20 year renewal of the Class 104c Facility Operating License No. R 130, Docket No. 50 607, for the University of California – Davis McClellan Nuclear Research Center (UCD MNRC) Training, Research, Isotope, General Atomics (TRIGA) nuclear reactor. By letter dated July 6, 2020, as supplemented, the licensee updated its LRA to reflect its decision to reduce the licensed thermal operating power level from 2.3 megawatt-thermal (MWt) to 1.0 MWt, and to eliminate pulsing capability and irradiation of explosive materials in the reactor tank.

During the NRC staff's review of the updated UCD MNRC LRA, questions have arisen related to the safety analysis report (SAR), technical specifications (TSs) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20188A371), and the environmental report (ER) (ADAMS Accession No. ML20238B993), submitted by letter dated July 6, 2020, for which additional information is needed to determine that there is reasonable assurance of adequate protection of public health and safety and that applicable regulatory requirements are met. These questions identify additional information needed for the NRC staff to continue its review and may become formal requests for additional information following the December 14 through December 18, 2020, regulatory audit.

Regulatory Basis and Applicable Guidance Documents

The LRA for UCD MNRC reactor is being evaluated using the appropriate regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), and the following guidance:

- NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," issued February 1996 System (ADAMS Accession No. ML042430055)

- NUREG-1537, Part 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," issued February 1996 (ADAMS Accession No. ML042430048)

- Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors," October 2009, (ADAMS Accession No. ML092240244).

- American National Standards Institute/American Nuclear Society, (ANSI/ANS)-15.1-2007 (R2013), "The Development of Technical Specifications for Research Reactors."

Attachment

The following questions refer to the UCD MNRC SAR (ADAMS Accession No. ML20238B984), unless stated otherwise.

General

Question G-1

Clarification of the Emergency Core Cooling System

- There is no discussion of what constitutes the emergency core cooling system (ECCS) in the facility. Even if it is not credited in the safety analysis, the system should be described somewhere if it still exists as part of the facility.

- On page 3-9, it states "Criterion 17: Electric Power Systems - An uninterruptible power supply (UPS) provides electrical power to the reactor console, DAC [Data Acquisition Computer], and translator rack during normal reactor operations. An additional emergency generator is provided to supply power to the Auxiliary Make Up Water System (AMUWS) and the reactor room exhaust fan (EF-1) should these systems be called upon to provide backup to the reactor ECCS system." **(This states that EF-1 is a backup to the ECCS.)**

- On pages 3-13 & 3-14, it states "Criterion 34: Residual Heat Removal - Calculations performed for loss of coolant show that an ECCS connected to the domestic water supply is sufficient to assure that fuel temperatures will not reach the safety limit even under loss-of-coolant conditions (Chapter 13). The AMUWS and the reactor room exhaust fan also provide a back-up capability to the ECCS system sufficient to provide the emergency cooling function should the domestic water supply also fail." **(This is inconsistent with Chapter 13 which states that an ECCS is not required. This states that ECCS is connected to the domestic water supply, which would imply ECCS consists of pumped flow. In addition, it states the exhaust fan is a backup to the ECCS.)**

- On page 3-14, it states, "Criterion 35: Emergency Core Cooling System - An emergency core-cooling system has been provided in the case of the unlikely probability that an accident such as a severe seismic event occurs which results in the instantaneous loss of all reactor coolant. Analyses presented in Chapter 13 show that sufficient capability resides in simply providing outside air to cool the core post LOCA." **(Is the exhaust fan part of the ECCS? Is it outside air during recirculation mode? There is no analysis in Chapter 13 which shows that simply providing air to cool the core is acceptable.)**

- On page 3-14, it states, "Criterion 36: Inspection of Emergency Core Cooling System - All components of the ECCS system are located in open spaces and are readily available for periodic inspection. Verification of the availability of the domestic water system is checked on a daily basis." **(This implies the ECCS consists of a pumped flow requiring a water supply.)**

- On page 6-1, it states "The subcomponent of 2.0 MW ECCS that is still required if the Exhaust Fan #1 (EF1). The uniquely small reactor room at MNRC results in a relatively small air volume to act as a thermal heat sink in the event of the complete instantaneous

LOCA. Therefore, in the event of a complete instantaneous LOCA EF1 would be required to operate so that fresh cool air is introduced to the reactor room to provide a heat sink for the decay heat of the reactor." **(This states that EF1 is part of the ECCS which is inconsistent with GDCs 17 and 34 which state that EF1 is a backup to the ECCS. Is EF1 really a part of the Engineered Safety Features (ESF) and not the ECCS?)**

- On page 9-16, Section 9.3.1, Auxiliary Make-Up Water System (AMUWS), states "The water storage tanks have enough capacity, if needed, to supply water to the reactor core area for approximately four hours at twenty gallons per minute as a backup supply to the ECCS." **(This says AMUWS is backup supply to the ECCS, however, per Chapter 6, there is no ECCS required.)**

Chapter 3 - Design of Structures, Components, Equipment and Systems

Question 3-1

On page 3-10, it states "Criterion 18: Inspection and Testing of Electric Power Systems - The primary power distribution system supplying commercial power to UCD/MNRC is maintained by electrical utility maintenance crews. Routine inspections of the systems are performed. The UCD/MNRC can tolerate a total loss of electric power with no adverse effects on the safety of the facility. There are no electrical power (distribution) systems designated as necessary to provide power to the UCD/MNRC during either normal or abnormal conditions." However, on page 3-9, it states "Criterion 17: Electric Power Systems - An additional emergency generator is provided to supply power to the Auxiliary Make Up Water System (AMUWS) and the reactor room exhaust fan (EF-1) should these systems be called upon to provide backup to the reactor ECCS system," which implies that electrical power systems are required. As stated on page 6-1, "… in the event of a complete instantaneous LOCA EF1 would be required to operate so that fresh cool air is introduced to the reactor room to provide a heat sink for the decay heat of the reactor." The statements for GDC 17 and GDC 18 seem inconsistent.

Question 3-2

On pages 3-13 & 3-14, it states "Criterion 34: Residual Heat Removal - Calculations performed for loss of coolant show that an ECCS connected to the domestic water supply is sufficient to assure that fuel temperatures will not reach the safety limit even under loss-of-coolant conditions (Chapter 13)." This is inconsistent with Chapter 13 in that the ECCS is not credited in the LOCA analysis.

Question 3-3

On page 3-15, it states "Criterion 50: Containment Design Basis - Under the conditions of a loss of coolant, it is conceivable that the temperature at the reactor room could **increase slightly** due to heating of the air flowing through the core. However, since the building is not leak tight, it will not pressurize from the heating of the air." In Section 13.2.3.2.2.1 (page 13 12), "Air Cooling," it states that the temperature in the reactor room could increase by 100°F which seems more than "increase slightly." In addition, can the equipment in the reactor room survive a temperature increase of 100°F?

Chapter 4- UCD/MNRC TRIGA® REACTOR

Question 4-1

Many of the figure numbers in the text do not seem to match the figure numbers in the caption. For example, on page 4-2, Figure 4.2 is referenced, but the nearby text seems to be describing Figure 4.1. In addition, some figure numbers in the captions are not sequential, i.e. the figure following Figure 4.26 is labeled as Figure 4.2, the figure following Figure 4.28 is labeled as Figure 4.3, etc.

Question 4-2

Figures 4.5 (page 4-8) and 4.6 (page 4-11) show that the TRIGA fuel elements contain a molybdenum spacer, but this is not discussed in the text. What is the purpose of this spacer?

Question 4-3

On page 4-9, Section 4.5.4.1 was referenced, but no such section exists. Should this be Section 4.6.4.1? This is repeated several times throughout the text.

Question 4-4

In Table 4.1 on page 4-10, U-235 enrichment is listed as ≤20%. Is there a lower-bound or nominal value for this number? In addition, per 10 CFR 110.2, "Definitions," low-enriched uranium means less than 20%, not less than **or equal** to 20%.

Question 4-5

On page 4-15, it states "The control system has been configured to provide for the excess reactivity needed for 1 MW operation 24 hours per day (including xenon override) and will be capable of providing a shutdown reactivity greater than 50 cents, even with the most reactive control rod in its most reactive position and moveable experiments in their most reactive position." Does this shutdown margin include positive reactivity that would be introduced by xenon production after a scram? If so, provide the analysis.

Question 4-6

On page 4-24, it states "If the reactor were to experience a complete LOCA the reactor would become subcritical even with all control rods withdrawn and the maximum licensed reactor excess reactivity." Provide analysis that support the statement.

Question 4-7

On page 4-37, it states "The UCD/MNRC fuel consists of U-ZrH with a H/Zr ratio between 1.6 and 1.7 and with 20 or 30 wt% enriched in ^{235}U to approximately 20% ^{235}U." Should this state "with 20 or 30 wt% uranium, enriched in"?

Question 4-8

On page 4-37, it states “The cladding is 0.020 in. thick stainless steel and has an inside diameter of 1.43 in,” and then goes on to calculate the stress as S = P · r / t resulting in a value of r / t of 36.7. This calculation appears to use the outer radius for the value of r. However, the equation for stress in a thin-walled cylinder uses the mean radius, which is 0.725 in. Therefore, r / t = 0.725 in / 0.020 in = **36.25**, and not **36.7**.

Question 4-9

How were power distributions shown in Figure 4.21 (page 4-42) and Figure 4.22 (page 4-43) obtained? Also, it appears that these power distributions were normalized differently.

Question 4-10

A fuel temperature transient following a pulse is described in Section 4.6.4.1.1 (page 4-41), Fuel and Clad Temperature During Pulsing. In this simulation, how does power change as a function of time, and are delayed neutrons considered?

Question 4-11

In the first paragraph on page 4-44, the conclusion is reached that a pulse resulting in a peak fuel temperature of 1,000°C will result in a peak clad temperature of 470°C. Then, the statement “Further analysis shows that this peak clad temperature is valid for a higher peak fuel temperature.” is made. What further analysis is being referred to, and to what higher peak temperature is this analysis valid?

Question 4-12

On page 4-44 it states, “Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad.” What temperatures need to be reached to induce such a gap in the fuel? What measurements are being referred to?

Question 4-13

On page 4-44 it states “This result is in agreement with experimental evidence obtained for clad temperatures of 400°C to 500°C for TRIGA® Mark F fuel elements.” Are these fuel elements comparable in design to the current fuel elements under consideration?

Question 4-14

Figure 4.25 (page 4-47) states that a flow velocity of 3 ft/s is assumed, however, on page 4-49 it is stated that a flow velocity of 1 ft/s is assumed. What flow rate was assumed for this calculation, and is this reasonable for the time scales being simulated? Based on the data provided in Table 4.13 (page 4-89) and Table 4.15 (page 4-95), it appears to the NRC staff that the steady-state coolant velocity in the hot channel at the top of the core should be v = m_dot / (ρA) = 0.0729 kg/s / (962.88 kg/m3 * 3.80E-4 m2) = 0.199 m/s = 0.654 ft/s. Clarify the steady state coolant velocity.

Question 4-15

On page 4-54, $\bar{c}$ is defined as the average gas concentration in the cylinder. What is this average performed with respect to (time? Volume? Axial length?)

Question 4-16

On page 4-56, it states that a 0.125 inch gap is assumed to exist at the top of the fuel element. This gap is not described in Section 4.2.1.1 (page 4-7), Fuel-Moderator Element. Is this part of the fuel element design?

Question 4-17

On page 4-56, sensitivity to parameters is investigated by changing the gap volume and the hydrogen temperature simultaneously. Should calculations have been done where the parameters were varied individually?

Question 4-18

On page 4-56, the fuel temperature is assumed to be invariant with radius. Why was this assumption made? What is the effect of this assumption?

Question 4-19

On page 4-59, it is stated that some amount of hydrogen will escape from the fuel rod through the cladding. How significant is this relative to the total amount of hydrogen that evolves from the fuel during a pulse?

Question 4-20

On page 4-60, it states that the prompt negative temperature coefficient mirrors that of reference 8.5 wt% uranium fuel, which contradicts earlier discussions of the negative temperature coefficient.

Question 4-21

On page 4-67, Figure 4.33 is referenced as showing the temperature coefficient for 20/20 fuel, however, the Figure 4.33 on page 4-69 shows a fuel map. Provide the correct figure.

Question 4-22

In Section 4.6.4.3 (page 4-69), Operating Core Configuration (OCC), the five fuel-followed control rods are described as having 20/20 type fuel as followers. Fuel-followers are not described in Section 4.6.4.4 (page 4-71), Limiting Core Configuration (LCC). However, on page 4-7, the first paragraph mentions that 30/20 fuel-followers may be loaded into the core. Is there a plan to use 30/20 fuel followers? If so, how will this affect control rod worth calculations and peaking factors?

Question 4-23

In Section 4.6.4.2.1 (page 4-67), Validation of MNRC MCNP Core Model, what MCNP cases are run to estimate the worth of control rods?

Question 4-24

On page 4-68, should the phrase “Highest Rod Worth Non-Secured Experiment” be “Highest Worth Non-Secured Experiment”?

Question 4-25

The caption for Figure 4.34 on page 4-70 starts with “OCC fuel burnup rate in percentages.” What is the unit of time for this burnup rate?

Question 4-26

In Figure 4.34, the central irradiation facility is pictured in the model. Will the facility always be loaded when the reactor is operating? If not, how will this change peaking factors pictured in Table 4.7?

Question 4-27

On page 4-71, it states “The purpose of this core evolution is to shift the highest fission rate fuel elements to coincide with the lowest burnup elements.” Does “coincide” mean move adjacent to? Or is it maximizing power in the lowest burnup elements?

Question 4-28

On page 4-73, the hot fuel rod is listed as 17.7 kW. It seems that this is just due to rounding, however, this is larger than the 17.69 kW shown on page 4-33.

Question 4-29

On page 4-77, the text implies that intra-rod power distributions depicted in Figures 4.41, 4.42, and 4.43 were tallied with MCNP. If so, what ranges of relative statistical uncertainty are associated with these tallies?

Question 4-30

Is the core configuration considered in Section 4.6.4.4 (page 4-71), Future Cores and the Limiting Core Configuration (LCC), limiting? Could higher peaking factors result from loading additional graphite reflector elements in the C-ring?

Question 4-30

On page 4-75, it states “By projecting 10 yrs of normal operation into the future, i.e. 12,000 MWhrs, the additional burnups on average of those 5 FFCRs, 14 30/20 type fuel elements, and 83 20/20 type fuel elements are 7.7%, 6.1%, and 5.4%, respectively.” What is the maximum burnup for each of these fuel element types? Why is the projection only performed 10 years into the future, when the license renewal will allow the reactor to operate for 20 additional years?

Question 4-31

In Figure 4.35 (page 4-71) and Figure 4.37 (page 4-73), power distributions at 1.1 MW are shown. Why is this power level used, when the safety setting is 1.02 MWt? If these power levels were tallied with MCNP, what level of relative statistical uncertainty is associated with this tally?

Question 4-32

How do the peaking factors shown in Table 4.7 (page 4-79) vary as control rods are inserted to different levels? For example, what would peaking factors be when control rods are inserted to the critical control rod position?

Question 4-33

Equations (18) and (19) (page 4-81) show $f(r_i, z_0)$ both on the left-hand-side and right-hand side. Equation (20) does not show how the peaking factor is calculated, since right-hand-side variables are continuous and not known.

Question 4-34

There are several references to tables in Section 4.7, Thermal and Hydraulic Design, that do not list table numbers (such as page 4-80) where is shows Table 4.X where X is missing.

Question 4-35

Provide the RELAP5 input and outputs used in Section 4.7 (page 4-77), Thermal and Hydraulic Design.

Question 4-36

On page 4-83, it states, "In these 20 axial nodal locations a modified cosine axial heat distribution has been applied, based on the results produced from the MCNP5 model." How is the cosine distribution modified to reflect the power distribution tallied in MCNP?

Question 4-37

Section 4.7.1 (page 4-82), Description of the RELAP5-3D Model, states "This model is based on a single-channel analysis assumed to represent the hottest channel via combination of smallest hydraulic geometry and highest-power element in the core." However, Section 4.7.3 (page 4-94), Steady State Results, states "The predicted steady state thermal-hydraulic performance of the MNRC OCC and LCC core configurations is determined for the reactor operating at 1.0 MW_{th}." How was a core configuration operating at 1.0 MW_{th} simulated with a single channel model? Why was 1.0 MW used when the scram setpoint from TS 3.1.1 is 1.02 MW?

Question 4-38

In Section 4.7.1.2 (page 4-84), Coolant Source (100), it states “The MNRC’s technical specification requires a minimum water column height above the top of the core to be 7.01 meters (23 feet),” however, this TS actually refers to the height of water in the tank, not above the core. What is the actual distance from the top of the core to the point where the water level is 23 feet above the bottom of the tank? This distance is going to be much smaller than the 23 feet of water assumed to be above the core in the calculations. The result will end up lowering the pressure at the core outlet and change all the MDNBR results. Provide drawings (similar to Figure 4.2 on page 4-4) that show dimensions. The core itself is modeled as being 0.711 meters high. The description of the safety plate in Section 4.2.6.3 (page 4-20), Safety Plate, suggests that there is at least an additional 18.25 inches (0.4636 m) below the core, implying that the actual height of the water column over the core should be at most 5.84 meters. Based on description of the central thimble in Section 10.4.1 (page 10-12), Central Irradiation Facility, there are less than 55 inches from the bottom of the tank to the top of the core which would result in 5.61 meters of water over the top of the core, at most.

Question 4-39

On page 4-86, the pitch is listed as 1.60 inches, which is inconsistent with the pitch listed on page 4-17, of 1.714 inches. Which is the correct pitch?

Question 4-40

In Table 4.13 on page 4-89, the flow area is listed as 3.80E-4 m^2, which contradicts the value of 3.3E-4 m^2 listed on page 4-86. What flow area was used in the RELAP5 calculations used to obtain the results shown in this section?

Question 4-41

In Table 4.13 on page 4-89, why do “Heated Diameter” and “Hydraulic Diameter” have different values? The wetted perimeter is the same as the heated perimeter in the subchannel shown in Figure 4.47 on page 4-86.

Question 4-42

In Table 4.13 on page 4-89, how are inlet and exit pressure loss coefficients derived from the values given in the text (on pages 4-88 and 4-89)?

Question 4-403

On page 4-91, it states “The fuel to clad contact gap that is created by material surface roughness is originally hydrided during manufacturing of TRIGA® fuel. As the U-ZrH fuel is burnt through its lifetime fission product gasses are released and migrate from the fuel lattice structure into the gap.” Is this the same process referred to on page 4-44, which states “Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad”?

Question 4-44

On page 4-92, it states “The outer gap coordinate (Node 22) is varied during this study.” What gap thickness was used for the results shown in Section 4.7.3 (page 4-94), Steady State Results? Are the results sensitive to this parameter? It appears from the data in Table 4.14 (page 4-92), Heat structure radial node lengths, that the fuel cladding thickness was changed to vary the gap width. Why was this done as opposed to varying the outer fuel radius?

Question 4-45

Table 4.14 (page 4-92 and 4-93), Heat structure radial node lengths, appears to have inconsistent information relative to other portions of the document. The table shows a cladding outer radius of 0.7375 in, however, it is shown as 1.5 in / 2 = 0.75 in on page 4-8, while Table 4.1 (page 4-10), SUMMARY OF FUEL ELEMENT SPECIFICATIONS, shows it as 1.47 in / 2 = 0.735 in. The table shows a fuel outer radius of 0.70275 in, however, Table 4.1 on page 4-10 shows a value of 1.43 in / 2 = 0.715 in. Table 4.1 on page 4-10 shows a cladding thickness of 0.02 in, while Table 4.14 shows values of either 0.03465 of 0.03445 (depending on the gap thickness).

Question 4-46

Section 4.7.3 (page 4-94), Steady State Results states “Groeneveld 1986, 1995, and 2006 [4.62] critical heat flux tables were used as the primary means for predicting margin to departure from nucleate boiling with the he Bernath correlation [4.61] provided as a qualitative reference for historic purposes.” Were the results for DNB computed directly by RELAP5, or were RELAP5 output parameters used to separately compute DNB?

Question 4-47

The figures shown on pages 4-95 through 4-97 have an inconsistent numbering scheme (the sequence goes 4.51, 4.52, 4.51, 4.52, 4.53).

Question 4-48

On page 4-95, the text describes coolant mass flux as being shown in Figure 4.52. Coolant mass flux is not shown in either figure labeled Figure 4.52. Coolant mass flow rate is shown in Figure 4.51 on page 4-95, is this what the text is referring to?

Question 4-49

Provide the following information (based on the acceptance criteria given in the listed Section of Chapter 4 of NUREG-1537, Part 2):

Section 4.5.1, Normal Operating Conditions
- Reactivity worth of fuel elements
- Dynamic reactivity parameters of instrumentation and control systems

Section 4.5.2, Reactor Core Physics Parameters
- Provide and justify uncertainties in the analyses
- Show methods used to analyze neutron lifetime, effective delayed neutron fraction
- Methods and assumptions for calculating the various neutron flux densities should be validated by comparisons with similar reactors.

- Uncertainties and ranges of accuracy for analyses requiring neutron flux densities, such as fuel burnup, and thermal power densities.

Section 4.5.3, Operating Limits

- Analysis and discussion of operational requirements for excess reactivity, as it pertains to:
    - Void coefficients
    - Xenon and samarium override

<u>Question 4-50</u>

Interim Staff Guidance on the Streamlined Review Process for License Renewal for Research Reactors (October 2009), Section 4.5.3, states "… the reviewer should confirm that the value of the [limiting safety system setting] LSSS protects the hot coolant channel in the core from burnout and protects the safety limit in the hot channel for all allowed core locations of the instrumented fuel element." The basis for TS 2.2.1 states "If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is near the center and mid-plane of the fuel element." How will this change if control rods are partially inserted, and the axial power profile is no longer symmetric?

<u>Question 4-51</u>

Provide the following documents referenced in the SAR:

4.23 Johnson, H. E., "Hydrogen Disassociation Pressures of Modified SNAP Fuel," Report NAA-SR-9295, Atomics International, 1964.

4.24 West, G. B., M. T. Simnad, and G. L. Copeland, "Final Results from TRIGA® LEU Fuel Post-Irradiation Examination and Evaluation Following Long Term Irradiation Testing in the ORR," GA-A18641, November 1986.

4.25 Baldwin, N. L., F. C. Foushee, and J. S. Greenwood, "Fission Product Release from TRIGA® LEU Reactor Fuels," GA-A16287, November 1980.

4.48 Marcum, W.R., et al., Steady-State Thermal-Hydraulic Analysis of the Oregon State University TRIGA Reactor Using RELAP5-3D. Nuclear Science and Engineering, 2009. 162(3): p. 261-274.

4.49 Marcum, W.R., B.G. Woods, and R. S.R., Experimental and Theoretical Comparison of Fuel Temperature and Bulk Coolant Characteristics in the Oregon State TRIGA® Reactor during Steady State Operation. Nuclear Engineering and Design, 2010. 240: p. 151-159.

4.50 Marcum, W.R., et al., A Comparison of Pulsing Characteristics of the Oregon State University TRIGA® Reactor with FLIP and LEU Fuel. Nuclear Science and Engineering, 2012. 171(2): p. 150-164.

4.59 Simnad, M., F. Foushee, and G. West, Fuel elements for pulsed TRIGA Research Reactors. 1975, General Atomics: San Diego, CA.

4.61 Bernath, L., A Theory of Local Boiling Burnout and Its Application to Existing Data. Chem. Eng. Progr., 1960. 30(56): p. 95-116.

4.62. Groeneveld, D.C., et al., The 2006 CHF Look-up Table. Nuclear Engineering and Design, 2017. 237(15-17): p. 1909-1922.

Chapter 6- ENGINEERED SAFETY FEATURES

Question 6-1

In Chapter 6.0, Engineered Safety Features, it states "Previous analysis has shown that an ECCS was not required for the UCD/MNRC, since at 1 MW even an instantaneous loss of the entire tank water would not have resulted in fuel temperatures which would have threatened the fuel clad." What previous analysis is being referenced in this statement? The analysis provided in Section 13.2.3.2.2.1 (page 13-12), Air Cooling, states that the continuous flow of air provides the ultimate heat sink for the decay heat of the core, however, it does not provide any results for fuel temperatures.

Chapter 7 - INSTRUMENTATION AND CONTROL

Question 7-1

On page 4-13, it states that control rod fuel followers may have either 20 or 30 weight percent uranium, but on page 7-19, it specifies that the weight percent of uranium in the fuel-follower is 20. Which is the correct number?

Question 7-2

Several figure number references appear to be off by 1 (e.g., Text on page 7-19 references Figure 7.10, but it seems the text is actually referring to Figure 7.9).

Question 7-3

On page 7-19 it states, "The transient rod is a sealed, 44.25 in. long by 1.25 in. diameter tube containing solid boron carbide as a neutron absorber and air as a follower. The absorber section is 21 in. long and the follower is approximately 23 in. long. The transient rod passes through the core in a perforated aluminum guide tube." What material composes the transient rod tube?

Chapter 10 - EXPERIMENTAL FACILITIES AND UTILIZATION

Question 10-1

In Section 10.4.1 (page 10-12), Central Irradiation Facility, it states "Once installed in the central cavity, the central thimble shall not be removed from the reactor core unless it is to be replaced with another facility of similar dimensions that has been analyzed to show how it affects the overall operation of the reactor (See Section 10.4.1.1)." Section 10.4.1.1 (page 10-12), Central Irradiation Facility (CIF-1), begins "The central irradiation fixture (CIF-1) consists of a graphite thimble plug and associated removable aluminum thimble plug insert positioned in the central irradiation facility (Figure 10.8)." In this description, does the central thimble include the central aluminum plug and central graphite plug? That is, can the graphite plug be removed from the core without performing any additional analysis? What kind of analysis will be performed if the central thimble is to be removed from the core?

Chapter 11- RADIATION PROTECTION AND WASTE MANAGEMENT PROGRAM

Question 11-1

In Section 11.1.2.1.5 (page 11-10), Production and Evolution of N-16 in the Reactor Room, it appears to reference Appendix A of the SAR as a source of the calculations done to show the dose results from N-16 plane source calculation. However, when reviewing Appendix A, the dose calculations cannot be found. Provide calculation for doses from N-16.

Question 11-2

10 CFR 20.1502, "Conditions requiring individual monitoring of external and internal occupational dose," requires monitoring of workers likely to receive, in one year from sources external to the body, a dose in excess of 10 percent of the specified limits. How does MNRC determine the badging criteria for individuals expected to receive exposures of radiation? Are there minimum administrative criteria to be met for being monitored?

Chapter 13 - ACCIDENT ANALYSIS

Question 13-1

In Section 13.1 (page 13-1), it states "Fuel temperature limits of 1100°C (with clad <500°C) and 930°C (with clad >500°C) for U-ZrH with a H/Zr ratio less than 1.70 have been set to preclude the loss of clad integrity (Section 4.5.4.1.3)." Section 4.5 of the SAR is Primary Coolant, and there is no Section 4.5.4.1.3 in Section 4.5 (page 4-24), Primary Coolant. Should this be Section 4.6.4.1.3? Note that Section 4.5.4.1.3 is also mentioned in Section 13.2.2.2.1 (page 13-6) and Section 4.6.2 (page 4-31).

Question 13-2

In Section 13.2.1.2 (page 13-2), Accident Analysis and Determination of Consequences, it states "This release fraction is developed in Chapter 4 and is based on the maximum measured fuel temperature (400 C) which corresponds to the average fuel temperature of the highest thermal output fuel element of the LCC core." What is the basis for the 400°C and where is it mentioned in Chapter 4?

Question 13-3

In Section 13.2.2.2.1 (page 13-6), Maximum Reactivity Insertion, it states "The quantity that captures this effect is the prompt negative temperature coefficient discussed in Section 4.5.4.2." There is no Section 4.5.4.2. Should this be Section 4.6.4.2?

Question 13-4

In Section 13.2.2.2.1 (page 13-7), Maximum Reactivity Insertion, the initial temperature, T_o, is set to 20°C [68°F], which is the nominal zero-power temperature. Is this temperature conservative in the calculation for the determination of the maximum reactivity insertion allowed? How are the results sensitive to the initial temperature assumption? In NUREG-1630, Safety Evaluation Report Related to the Issuance of a Facility Operating License for the Research Reactor at McClellan Air Force Base, Dated August 13,1998 (ADAMS Accession No. ML053210295), it appears temperatures of 35°C and 257.2°C were used.

Question 13-5

In Section 13.2.2.2.1 (page 13-6 and page 13-7), Maximum Reactivity Insertion, states equations were solved numerically using simple finite difference techniques and by iteration. Were these done by hand, or with some code? Are the calculations documented anywhere besides the SAR? If so, provide the calculations.

Question 13-6

In Section 13.2.2.2.1 (page 13-7), Maximum Reactivity Insertion, it states "The value of PF was selected to be 4.86 which is significantly larger than the 3.69 LCC is capable of producing." What is the basis for both 4.86 and 3.69? These values were not found anywhere in the SAR.

Question 13-7

In Section 13.2.2.2.1 (page 13-7), Maximum Reactivity Insertion, the value for B is 0.007, while it is 0.0075 in Section 4.6.4.2.1 (page 4-67) and 0.0076 in Section 13.2.2.2.2 (page 13-9). Why are different values used?

Question 13-8

The conclusion, in Section 13.2.2.2.1 (page 13-9), Maximum Reactivity Insertion, is the maximum accidental reactivity insertion that could occur with no risk of fuel damage is $1.92. This value is the same as that from the previous NRC safety evaluation report (SER) (See page 13-5 of NUREG-1630). Is it just a coincidence that a different power level, fuel, heat capacity and prompt negative temperature coefficients would lead to the exact same value of $1.92. Note that the original application, dated October 23, 1996, (ADAMS Accession No. ML20129H791), computed a value of $2.12, so there must have been a supplement between the original application and the NRC SER that changed the $2.12 to $1.92. Provide the detailed calculations that determined the current $1.92.

Question 13-9

In Section 13.2.2.2.1 (page 13-7), Maximum Reactivity Insertion, Table 13-4 presents equations for prompt negative temperature coefficient for different fuel types. Where did the equations come from and why does Figure 13.1 stop at 800°C when the calculations go up to 1,100°C?

Question 13-10

In Section 13.2.2.2.2, Uncontrolled Withdrawal of a Control Rod, it states "The maximum single rod worth for the reference loadings of Section 4.5.5 is ~$2.70, but a rod worth of $3.00 for the 5 fuel followed control rod and $2.50 for the transient rod was used here to allow for reasonable variations about the reference loadings." Section 4.5 is Primary Coolant, and there is no Section 4.5.5. In addition, the value of $2.70 was not found in the SAR. Provide the correct reference and confirm that $2.70 is the maximum single rod worth.

Question 13-11

In Section 13.2.2.2.2 (page 13-9), Uncontrolled Withdrawal of a Control Rod, it states "The SCRAM set point is 1.1 MW and a delay of 0.5 seconds is assumed between the set point being reached and the initiation of the controls dropping into the core." TS 3.2.1.b states "The scram time measured from the instant a signal reaches the value of a limiting safety system setting to the instant that the slowest control rod reaches its fully inserted position shall not exceed one (1) second." What is the basis for the 0.5 second delay time? Is there any data available to determine how much of the one second is delay time versus the actual rod drop time? In addition, the SCRAM setpoint in TSs is 1.02 MW. Was 1.1 MW used for conservatism?

Question 13-12

In Section 13.2.2.2.2 (page 13-9), Uncontrolled Withdrawal of a Control Rod, it states "The reactivity worth was assumed to be linear along the length of the active 15 inches of their travel." In the limiting accident scenario, after $1 of reactivity is inserted, power rises rapidly and trips the power level scram almost instantaneously. Then, any inserted reactivity greater than $1 is only a function of the scram delay and the reactivity insertion rate. In the middle of their travel, differential control rod worth is higher than the average control rod worth. If the scram occurs when control rods are in the middle of their travel, what will be the total reactivity insertion?

Question 13-13

In Section 13.2.2.2.3 (page 13-10), Uncontrolled Withdrawal of **All** Control Rods, it states "Beginning at power level of 100 W **a single rod** being withdrawn until the SCRAM set point is reacted takes 1.23 seconds plus an additional 0.50 seconds for control drop to initiate. This corresponds to a reactivity insertion of $1.415." and "Beginning at power level of 1.0 MW **a single rod** being withdrawn until the SCRAM set point is reacted takes 0.13 seconds plus an additional 0.50 seconds for control drop to initiate. This corresponds to a reactivity insertion of $0.515." Both of these statements state **a single rod** being withdrawn, however, this section is for **all rods withdrawn**. Clarify if this should have been all rods withdrawn, three rods withdrawn, or something else. For the multiple withdrawal of rods, what is the assumed rod worth?

Question 13-14

In Section 13.2.2.2.4 (page 13-10), Beam Tube Flooding or Removal, it states "It has been estimated that the worth of one flooded beam tube is about $0.25." What is the basis for this estimate? Is it sensitive to loading pattern, such as 30/20 elements being loaded in the outer ring or control rod position (such as asymmetric control rod insertion inducing a flux tilt toward a flooded beam port)?

Question 13-15

In Section 13.2.2.2.5 (page 13-10), Metal-water Reactions, it states "Water quench tests on TRIGA® fuel have been conducted to fuel temperatures as high as 1200°C without significant effect. Since the operating temperatures at 1 MW do not approach this temperature, this effect does not represent a safety risk." While operating temperatures do not approach this temperature, are there any transients or accidents that can result in the fuel exceeding this temperature? Given that the SAR does not provide fuel temperature calculations for all scenarios (e.g., LOCA, loss of coolant flow), what is the highest fuel temperature that can be obtained during an accident?

Question 13-16

In Section 13.2.3.2.2.1 (page 13-12), Air Cooling, it states “As with all TRIGA reactors operating under 1.5 MW (NUREG/CR-2387) no emergency core cooling is required in the event of instantaneous LOCA.” NUREG/CR-2387, “Credible Accident Analyses for TRIGA-and TRIGA-Fueled Reactors,” dated April 1982 doesn’t state this. It looked at a specific case at 1.5 MW and an instantaneous loss of the pool water and concluded that radiative loss of the core heat would be enough to ensure cladding integrity. However, given that “Unlike most other TRIGA reactors the MNRC has an unusually small reactor room due to its purpose built nature,” what is the basis that no ECCS is needed? There are no calculations provided that demonstrate air-cooling is enough (i.e., show peak cladding temperature).

Question 13-17

For Section 13.2.3.2.2.1 (page 13-12), Air Cooling, there is only a qualitative description of how the heat from the core will be removed. What calculations were done to determine that the 800-cfm exhaust flow is sufficient? What is the peak fuel temperature during events that require air cooling? Does it always remain below 1,100°C? In addition, this section discusses air flow in the reactor room during “normal operation.” How does operation in “recirculation” mode change the cooling capability?

Question 13-18

Overall, Section 13.2.4 (page 13-19), Loss of Coolant Flow, does not provide any quantitative analysis for local conditions, rather it only discusses the bulk tank temperature. As noted on page 13-11 of NUREG-1537, Part 2, the analysis should show that the peak fuel temperature does not reach an unacceptable value. For Section 13.2.4.3 (page 13-20), Localized Loss of Coolant Flow, it assumes an operator will SCRAM the reactor if a foreign object or debris is dropped into the tank, however, it does not consider the possibility that debris enters the tank without an operator noticing. This section should assume that some debris does enter the tank and blocks fuel coolant channel(s) and provide analysis of heat transfer around the area of blockage to demonstrate that the peak fuel temperature does not reach an unacceptable value.

Question 13-19

In Section 13.2.6.2 (page 13-23), Accident Analysis and Determination of Consequences, it states “A specific limitation of less than $1.00 on the reactivity of individual moveable experiments placed in the reactor tank has been established and is safe because analysis has shown that pulse reactivity insertions of $1.75 in the UCD/MNRC reactor result in fuel temperatures which are well below the fuel temperature safety limit of 930°C (Section 13.2.2).” However, Section 13.2.2 established that a reactivity insertion of $1.92 will keep the fuel temperature less than 1,100°C, not that a $1.75 reactivity insertion will keep fuel temperatures under 930°C.

Question 13-20

Section 13.2.7, Loss of Normal Electrical Power, states that emergency backup power systems are not required. However, page 3-9 states “Criterion 17: Electric Power Systems - An uninterruptible power supply (UPS) provides electrical power to the reactor console, DAC, and translator rack during normal reactor operations. An additional emergency generator is provided to supply power to the Auxiliary Make Up Water System (AMUWS) and the reactor room exhaust fan (EF-1) should these systems be called upon to provide backup to the reactor ECCS system." What will power EF-1 if the emergency backup power system is not required?

Question 13-21

In Section 13.2.7.2 (page 13-25), Accident Analysis and Determination of Consequences, it states "Since the UCD/MNRC does not require emergency backup power systems (see Chapter 6) to safely maintain core cooling, there are no credible reactor accidents associated with the loss of electrical power." Chapter 6 does not discuss power systems and this seems inconsistent with page 3-9 which states "Criterion 17: Electric Power Systems - An uninterruptible power supply (UPS) provides electrical power to the reactor console, DAC, and translator rack during normal reactor operations. An additional emergency generator is provided to supply power to the Auxiliary Make Up Water System (AMUWS) and the reactor room exhaust fan (EF-1) should these systems be called upon to provide backup to the reactor ECCS system."

Question 13-21

While not caused by a loss of normal electrical power, does a loss of normal electrical power need to be considered for a LOCA? Since EF1 is required during a total loss of coolant, can it operate during a loss of normal electrical power?

Question 13-22

Provide the following documents referenced in the SAR:

13.5 Research Reactor Core Conversion Guidebook, IAEA-TECDOC-643, April 1992; reprinted as UZR-27 by General Atomics.

13.16 10 CFR 50.59 Safety Analysis of Explosive Limits for Radiography Bays 1, 2, 3, and 4, McClellan Nuclear Radiation Center, Sacramento, CA., 1996.

13.17 Southwest Research Institute, "Safety Analysis To Determine Limiting Criteria for Explosives in Bay 3 of the McClellan Nuclear Radiation Center," September 1995.

Accident Dose Audit Information Needs

Question 13-1 (Accident Dose)

Chapter 13, Section 13.2.1.2 (page 13-3), Accident Analysis and Determination of Consequences, and Appendix B (page B-4), Radiological Impacts of Accidents, states, in part, for the maximum hypothetical accident (MHA): "Dose conversion factors from FGR 13 (ICRP 60/70 series) were used to along with a breathing rate of 4.17e-4 m^3/s to calculated various doses to the receptors of interest (1.5 m above ground level)."

The NRC's radiation protection regulations in 10 CFR Part 20, "Standard for Protection against Radiation," are based on Federal Guidance Reports (FGRs)-11/12 (the International

Commission on Radiological Protection (ICRP)-26/30 dose methodology and dose coefficients). Proposed use of dose methodologies and dose coefficients other than ICRP-26/30 requires an exemption to 10 CFR Part 20.

Typically, dose receptors are considered at 1 m above ground level to estimate calculations of representative whole-body doses rather than at 1.5 m above ground level.

Appendix B of 10 CFR Part 20 states that $2x10^4$ ml is the volume of air breathed per minute at work by "Reference Man" under working conditions of "light work."

Clarify the proposed use of the ICRP-60/70 dose methodology and dose coefficients. Provide the basis for the breathing rate of 4.17e-4 m^3/s and the dose receptors at 1.5 m above ground level used to calculate the MHA doses at the locations of interest.

Question 13-2 (Accident Dose)

Chapter 13, Section 13.2.1.2 (page 13-3), Accident Analysis and Determination of Consequences, for the MHA and Appendix B, Section B.1 (page B-4), Maximum Hypothetical Accident (MHA), state, in part, that the “HotSpot [computer code] was used to determine the worst case radiation dose impact to the public for a variety of atmospheric stability classes.” Provide the HotSpot computer code calculations for the NRC staff’s verification of the MHA doses.

Question 13-3 (Accident Dose)

Chapter 13, Section 13.2.3.2.2.2 (page 13-13), Ground Water Contamination, provides an equation (Eq. 7) for the penetration (delay) time to calculate radionuclide concentrations in the reactor tank water (7,000 gal assumed for release) reaching the ground water which are compared to the liquid effluent concentration limits in Table 2, Column 2 of Appendix B to 10 CFR Part 20. Using Eq. 7 with the parameter values given results in a different delay time (other than 36 hours) which results in different radionuclide concentrations in Table 13-5, “Predominant Radionuclides in Primary Coolant at Equilibrium and Upon Reaching Ground Water” on page 13-14. Clarify how the delay time of 36 hours is calculated using Eq. 7 with the values given.

Question 13-4 (Accident Dose)

Chapter 13, Section 13.2.3.2.2.3 (page 13-15), Radiation Levels from the Uncovered Core, for the Loss of Coolant Accident (LOCA) and Appendix B, Section B.3 (page B-8), Radiation Dose Rate from the Core Following a Loss of Coolant Accident, state, that the MCNP (Los Alamos National Laboratory Monte Carlo N-Particle (MCNP)) computer code is used to model the MNRC reactor and facility to calculate dose rates for “an individual directly over the core standing in the reactor room, inside the reactor room but not in direct line-of-sight of the core, just outside the reactor room, inside the control, at the MNRC fence line, in the closes building, and the closest inhabited building.” Provide the MCNP computer code calculations for the NRC staff’s verification of the LOCA doses.

Question 13-5 (Accident Dose)

Annual average atmospheric dispersion (X/Q) factors and lateral and vertical diffusion coefficients used to calculate the MHA doses with the design wind speed of 1 m/s at locations of interest are not provided in Chapter 13 and Appendix B. The NRC staff notes, however, that atmospheric dispersion information is provided on X/Q factors and lateral and vertical diffusion coefficients with the wind speed of 3.4 m/s to calculate routine (non-accident) doses to members of the public at locations of interest in Chapter 11, "Radiation Protection and Waste Management" and Appendix A, "Radiological Impact of Ar-41, N-16, Fission Products and Activated During Normal Operations" of the SAR. Provide the X/Q factors and lateral and vertical diffusion values used to calculate the MHA doses at the locations of interest.

<u>Question 13-6 (Accident Dose)</u>

Chapter 13, Section 13.2.3.2.2.3 (page 13-16), Radiation Levels from the Uncovered Core, appears to be missing a text discussion for Table 13-7, "Dose Rates Above the MNRC Reactor After a Loss of Pool Water Accident Following 1 MW Operations." Add text discussion for Table 13-7 (replace "Table B-6" with "Table 13-7" in the text above Table 13-7).

<u>Question 13-7 (Accident Dose)</u>

Appendix B, Section B.2 (page B-7), Single Element Cladding Failure in Water, provides an equation (D = 6 CEN) to assess more "realistic accident scenarios" in which it is "assumed that the pool water remains in the reactor tank (thus lowering the halogen dose significantly) and that the cladding failure occurs 24 hours after reactor shutdown" and "since most of the halogens will be retained in the primary coolant water, the majority of the activity will end up in the demineralizer resin beds." Two of the parameter values given in the equation include: "E = Energy of source in MeV = 1" and "N = Number of photons/dis = 1." However, Chapter 11, Table 11-5 (page 11-14), Representative Radioactive Sources for the UCD/MNRC," also identifies Co-60, contained in 6 resin bottles at a volume of 2 ft^3 each, in which E = 1.25 MeV and N = 2 photons/dis. Clarify how the dose rate at 1 ft is calculated for a demineralizer resin bed. Provide the basis for all parameter values (C, E and N) used in the equation and the point source geometry assumed.

<u>Chapter 14 Technical Specifications</u>

<u>Question 14-1</u>

Section 4.1 (page 4-1), Introduction, states, in part, the following:

> MNRC has not operated routinely above 2 MW nor has the reactor routinely pulsed for more than a decade. For the foreseeable future the MNRC will primarily function to support commercial and research neutron radiography and education/outreach programs. These programs can be accomplished by 1 MW single shift operations without pulsing. In order to operate the MNRC reactor with the largest operational safety margins as possible the reactor is no longer operated in pulse or square-wave mode.

However, in Section 7.1.2.5 (page 7-7), Reactor Operating Controls, states that the UCD/MNRC reactor can be operated in four modes: manual, automatic, square wave, and pulse. UCD notes that square wave and pulse mode are no longer utilized at MNRC.

Therefore, it's the NRC staffs' understanding that UCD is requesting a license to operate the MNRC in steady-state mode of operation only. Explain how both pulse and square-wave mode of operation is disabled/prevented. Further, explain why the proposed TS contain definitions, specifications, and bases related to pulse and square-wave mode of operation.

Question 14-2

TS 3.4.b, Reactor Room Exhaust System, it states “The reactor room exhaust system shall be operable within one half hour of the onset of a Loss of Coolant Accident.” There is no discussion of the half hour delay in the accident analysis in Section 13.2.3.2.2.1 (page 13-12), Air Cooling.

Question 14-3

The TS 4.4, Reactor Room Exhaust System, is used to verify the reactor room exhaust system is maintaining a negative pressure in the reactor room. This is for control of radiation exposure due to airborne sources. However, the exhaust fan is also credited for cooling after a LOCA. Is the requirement to maintain negative pressure enough to assure that it can meet the cooling requirements for a LOCA? Should there be a flow rate requirement?

Question 14-4

Specification 5.3.1, Reactor Core, Core and Other Variations, items 3 and 4 are identical. Item 3 should be “**20**/20 fuel may be used in any position in Rings C through G.”

Question 14-5

Specification 5.3.1, Reactor Core, Core and Other Variations, item 6 states “No single element may be operated at a power level above **17.7** kW (calculated) at a steady state power level of 1.0 MW.” While it’s a small difference, Section 4.6.3 (page 4-33), Design Criteria – Operating Core Configuration (OCC), Limiting Core Configuration (LCC), Planned Future Operating Core Configuration, and End of Life Planned Future Operating Core Configuration, states **17.69** kW.

Question 14-6

Specification 3.1.1 states, in part, “For the purpose of testing the reactor steady-state power level scram, the power shall not exceed 1.03 MW.” Explain if the reactor power level safety channels are tested during steady-state operations.

Environmental Report Data Needs

Question ER-1

Section 2.2.3 of the ER states, in part, that radiologically contaminated water may be encountered in the radiography bays and the men's washroom and that “the radiography bays have a drain system that leads to a sump in Bay 1.” The report further indicates that any water collected in the sump is pumped into an above ground liquid storage tank and further that the decontamination shower drains into the storage tank (also referred to as the “retention tank” by facility staff). However, information developed by the NRC staff based on observations made and discussions held with facility staff during the site audit indicate differing potential leak pathways for contaminated water as follows:

- Posted signage indicate that both the sink and the decontamination shower in the washroom drain to the sump system in Bay 1;
- Bay 1 is the only radiography bay that has a drain system and that there are no floor drains in any other bays that would convey leakage to the sump in Bay 1;

- The drain sump in Bay 1 could also receive overflow or leakage from the reactor tank (described in Section 3.3.1 of the ER) through a drain valve in the concrete shield and located on the wall opposite the sump in Bay 1;
- Any contaminated water collected in the Bay 1 sump would be conveyed to the above-ground storage tank (retention tank), which has no piped connection to the sanitary sewer system; and
- The retention tank is not piped to the sanitary sewer system.

Provide a brief summary description that confirms or clarifies this information, as appropriate.

<u>Question ER-2</u>

As referenced above and as described in Section 2.2.3 of the ER, an above-ground storage tank is used at the facility to prevent the accidental release of radiologically contaminated water to the environment. However, the storage capacity of this tank is not specified. Provide the volumetric capacity of the facility's retention tank located on the north side of the MNRC and specify the capacity of the concrete secondary containment structure in which the tank is located.

<u>Question ER- 3</u>

Section 2.2.3 of the ER indicates that in the event any water enters the facility's storage tank/retention tank, the water will be analyzed for radioactive materials. If any materials are found, Section 2.2.3 states that the water will be disposed of as discussed in Chapter 11 of the SAR. Section 4.2.12.2 of the ER and Section 11.1.2.2 (page 11-11), Liquid Radioactive Sources, of the SAR indicates that non-routine radioactive liquid waste is processed to a solid waste form on site and would be disposed of with other solid wastes.

- Clarify whether analytical samples would be collected from the sump in Bay 1 or from the retention tank directly;
- Describe how contaminated water contained in the retention tank would be retrieved, processed into a solid waste form, and disposed of;
- State the last occurrence (date) when radiologically contaminated water from the storage tank was conveyed to the sanitary sewer and indicate the volume(s) disposed and the activity levels of the liquids;
- State the last occurrence (date) when liquid waste in the retention tank was processed into solid waste for disposal, and indicate the volume(s) processed and disposed; and
- As discussed for larger volumes of liquid wastes (maintenance operations) described in Section 4.2.12.2 of the ER, state the last occurrence (date) when radiologically contaminated water associated with facility maintenance was conveyed to the sanitary sewer and indicate the volume(s) disposed, the activity levels of the liquids, and describe the sources of the waste.

<u>Question ER- 4</u>

Section 3.3.1 of the ER describes leak detection for the reactor tank. State whether there have been any known leaks (inadvertent release) of radiologically contaminated water to the subsurface from the facility. If so, provide a description of the leak(s), including dates and how discovered. Describe the type and results (include dates) of any environmental characterization or subsurface surveys conducted to characterize any leaks.

Question ER-5

Section 11.1.8 (page 11-52) of the SAR describes MNRC's environmental monitoring program and references the quarterly water quality monitoring conducted at off-site Well 54 (Site 42), as reported in the MNRC's annual reports. Provide a brief description of this well and include the depth and monitored geologic strata.

Question ER-6

As discussed in Section 11.2 (page 11-54), Radioactive Waste Management, of the SAR, MNRC Health Physics Procedure MNRC-0029-DOC is referenced as the procedure that addresses handling, storage and disposal of radioactive waste. To further describe waste minimization for the facility, provide procedure(s) that will describe how rad and non-rad waste generation will be reduced to the maximum extent possible.