DCS-016

SEP 2 1981

Docket No.: 50-170

Captain Paul E. Tyler, Director
Armed Forces Radiobiology Research
Institute
Defense Nuclear Agency
Bethesda, Maryland 20014

Dear Captain Tyler:

By letter dated July 10, 1981 you transmitted to the Nuclear Regulatory Commission a Safety Analysis Report (SAR) for your TRIGA reactor which you operate under NRC license R-84.

In accordance with your request, we are reviewing this report in connection with your application for renewal of your facility operating license.

Questions have arisen for which we require answers before we can complete that review. Therefore, please provide written responses to the enclosed list of questions within 15 days of receipt of this letter.

If you have any questions concerning this matter, please contact your Project Manager, Robert E. Carter on (301) 492-8106.

Sincerely,

ORIGINAL SIGNED BY:

James R. Miller, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
List of Questions

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cc w/enclosure

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*Through deposit in the Nuclear Regulatory Commission's internal mail system.

QUESTIONS ON THE AFRRI SAR

MAY 12, 1981

1. Describe the radiation protection staff. Identify the number, level, responsibilities of personnel, and the lines of communication.

2. Outline the minimum qualifications (training and/or previous experience) for each of your Health-Physics-related positions.

3. Describe any Health Physics training for non-Health Physics staff. If possible, provide a topic outline of the course, indicating the level and duration of each course.

4. Summarize your general radiation safety procedures. Identify the minimum frequency of survey, action points, and appropriate responses.

5. Describe the program to ensure that personnel radiation exposure and releases of radioactive material are maintained at a level that is "as low as reasonably achievable" (ALARA).

6. For your fixed-position radiation monitors, gas monitor, and effluent particulate monitor, specify the type of [illegible]ors and their efficiencies and the operable ranges.

7. For the fixed-position monitors, describe the methods and frequency of instrument calibrations and the routine operational checks.

8. For monitors that are alarmed, specify the alarm set-points and indicate the staff response to each alarm.

9. Identify the type, number, and operable range of all of the portable Health Physics instruments. Specify the frequency and methods of calibration.

10. If you anticipate that additional or specialized instrumentation may be readily available from other NNMC facilities, indicate the type, number, and range of the available equipment.

11. Describe your personnel monitoring program, including bio-assay and *in vivo* counting, if used.

12. Identify any administrative exposure limits and the anticipated actions if these levels are exceeded. Also, identify the operational constraints that are placed on personnel entering potential radiation/high radiation or contaminated areas.

13. What is the Health Physics review and exposure control of one-of-a-kind, short-term, low-to intermediate-risk tasks, such as simple but nonroutine maintenance activities and one-shot experimental measurements? If Special Work Permits (SWPs) are used for these events, please discuss the applicable requirements, limitations, and approvals.

14. Are the check valves in the purification and cooling system regularly checked for leakage?

15. Provide a summary of the AFRRI annual whole body radiation exposures (the number of persons receiving total annual exposure within the designated exposure intervals; that is, less than 0.5 rem, 0.5 to 1.0 rem, and so on) for past 10 years of operation.

16. What type of neutron dosimeter is used for personnel dosimetry? Describe briefly the neutron calibration of the dosimeter.

17. What checks are made during the readout of personnel dosimeters to assure the proper operation of read-out device?

18. Describe the radwaste handling systems and the applicable procedures, including methods for monitoring and measuring activity levels in radioactive waste before final disposal.

19. What are the technical specifications of the conductivity cells used to monitor the water demineralizer?

20. What functional tests are conducted on the reactor room ventilation isolation system and the overpressure relief damper? How frequently are these tests performed?

21. Describe the procedures for monitoring and changing the filters in the ventilation and water purification systems.

22. List and discuss the facility components that are on emergency back-up power. Describe the test and maintenance schedule for the back-up power supplies.

23. How many used fuel elements are in the pool storage racks? What problems will arise if the water level drops? If there is a total loss of coolant, how will the fuel elements be moved to repair the leak? Include the estimated exposures of operations personnel. (IV-14)

24. Describe the techniques and instrumentation used to insure the accurate determination of the reactor power level in both steady-state and pulse modes. Include the calibration methods and the frequency of calibration in the description.

25. How is the reactor tank constant measured?

26. Consider the overhead crane used in the regular operations.

    a. How often and by whom is it proof loaded?
    b. What is the safety factor between the proof loading and (1) the working load rating? (2) the maximum anticipated load?

27. Are there formal plans to place instrumentation in the ventilation system so that draw-fan failures can be readily detected? If so, describe; if not, justify.

28. Describe the techniques and procedures used to document reactor facility configuration, and changes in configuration.

29. Are the thermocouple channels that are used to measure the fuel element temperature calibrated? How often are they calibrated?

30. Under what conditions is operation of the ^{16}N diffuser system required?

31. Are the CCTV cameras routinely used for surveillance of the exposure rooms before reactor operation? Explain

32. What water, soil, and vegetation samples are taken from the area surrounding the liquid waste tanks?

33. What measurements are made to determine that the reactor areas exhausted into the AFRRI stack are maintained at a negative pressure? (See Fig. 3-5)

34. Describe the procedures and techniques used to prevent an inadvertent release of the contents of the radioactive liquid waste tanks to the public sewer.

35. Provide details of the communication system(s) linking the operator in the control room with the personnel operating the access doors to the exposure rooms.

36. How do you insure that any leakage from the primary cooling water circulating and demineralizing system goes into the hot water drain system?

37. How are the reactor room exhaust dampers checked for leakage when they are closed?

38. How are the "warm" and "hot" liquid waste tanks protected from corrosion? Who is responsible for the operation, inspection, and service of this system? Are the warm waste and hot waste storage tanks routinely inspected for leaks? For structural integrity?

39. Consider the effluent air monitoring systems.

    a. Describe the air particulate sampling probe in the reactor effluent line.
    b. What is the flow rate in the effluent line at this point and what is the sampling rate?
    c. Describe the gas sampling probe at the top of the effluent stack.

40. Are all of the operable thermocouples in each instrumented fuel element tied into the scram logic? Describe this portion of the scram system in detail.

The following questions are based on Chapter 6 of the 1981 Safety Analysis Report.

41. P. 6-6. You obtain a value of 0.08 MeV/(cm^3.s) for S_γ. We believe that it should be 0.18 MeV/(cm^3.s). Explain and verify the subsequent computations.

42. P. C-8, Table 2. For postulated critical excursions the Regulatory Guides recommend the consideration of more nuclides than you consider in your postulated accident. Justify your reasons for considering only the nuclides which you have.

43. P. 6-13, Para. 3. Justify the assumption that 99.8% of the iodine dissolves in the water?

44. P. 6-16, Eq. 3. Conversion factors are needed to yield "rad."

45. P. 6-17, Eq. 4. If BR = 3.47 x 10^{-4} m^3s is the breathing rate for Standard Man, it should be so stated.

46. P. 6-7, last paragraph. How did you obtain the ^{16}N activity of 1.2 x 10^{-2} $\mu Ci/cm^3$? Are the units correct?

47. Please provide a copy of the document stating the administrative policy that requires that a fuel element not be removed from the pool for at least 2 weeks following its use in the core. (p. 6-2)

48. P. 6-3, last sentence. Have there been experiments to check the 550° C "maximum average" fuel temperature for insertion of 2.8% $\Delta k/k$? How can you be certain that the fuel temperature will not exceed the technical specifications limiting safety system setting of 600° C for fuel temperature?

49. P. 6-5, Eq. 1. It is stated that Eq. 1 gives the activity production assuming a saturation condition. We believe that the equation is incorrect. Explain the presence of t in the numerator following Σ_a, and clarify the meaning of "saturation condition."

50. P. 6-7, Para. 2. We believe that the macroscopic 16O (n,p)16N cross section of oxygen in water is incorrect. We believe that it is 6.2×10^{-7} cm^{-1}. Please verify this and the following calculations that depend on the cross section. Furthermore, we remind you that the reaction is not produced by thermal neutrons.

51. P. 6-10, last sentence. Define thermal ratcheting.

52. P. 6-2, Para. 2. Please explain the significance of the distance of "25 meters from the AFRRI facility." What is the specific definition used in this document for the "AFRRI facility"?

53. P. 6-3, Sec. 6.2.2, Para. 2. Unless justified, please use consistent nomenclature for reactivity changes.

54. P. 6-3, Sec. 6.2.2, Para. 3. We believe that General Dynamics divested themselves of General Atomics before 1975. Please change as appropriate.

55. P. 6-3, Sec. 6.2.2, Para. 4. Please clarify the meaning of "a maximum average fuel temperature." If this means the maximum temperature in an average fuel element, please justify why you have not evaluated the maximum fuel temperature at any location in the reactor core.

56. P. 6-4, Sec. 6.2.3, Para. 1. Please clarify the meaning of "slightly critical."

57. P. 6-5, Sec. 6.2.4, Para. 2. What fuel loading is assumed in order to obtain 1×10^{13} for the average thermal neutron flux density? I obtain 0.6×10^{13} for your core. Please explain.

58. P. 6-5, last paragraph. It is stated that the assumed conditions represent "a highly conservative assumption for this facility". Please explain why you consider these conditions to be "highly conservative", and give a quantitative estimate of the magnitude.

59. P. 6-9, first line. Please justify the use of a distance of 25 meters from the "AFRRI facility."

60. P. 6-9, Sec. 6.3.1, Para. 1. Please discuss the radiological consequences of the demineralizer resins collecting all of the radionuclides from a fuel element cladding failure.

61. P. 6-9, Sec. 6.3.1, Para. 3, last line. "Reactivity" presumably refers to chemical reactions. Since "reactivity" is also a nuclear term, it is suggested that you clarify.

62. P. 6-10, Sec. 6.3.2.1, last sentence. Please clarify the meaning of "previously used aluminum cladding".

63. P. 6-11, Sec. 6.3.2.1, last sentence. Since the prior operating conditions were not given for the examples of failed clad fuel elements, it is not clear that the conclusion "would not constitute an undue risk..." is warranted under all conditions of possible operations. Please justify. Are these measurements consistent with the computations, paragraph one, page 6-13? Discuss.

64. P. 6-12, Sec. 6.3.2.2, Calculated Fission Product Inventory, Para. 1, Ultimate and penultimate sentences; please explain the significance to this computation of the terms "prompt fission" and "prompt inventories".

65. P. 6-15, Sec. 6.3.3, last paragraph. Please justify using a distance of 20 meters from the "AFRRI facility".

66. P. 6-19, reference 2. In addition to the internal General Atomics document, the following reference treats much of the same information, and in addition was published in a publicly available refereed journal: M. T. Simnad, F. C. Faushee, and G. B. West, Fuel Elements for Pulsed TRIGA Research Reactors, Nuclear Technology, 28, 31, (1976).

67. P. 6-12, Sec. 6.3.2.2, Determination of Fission Products in Gap, Para. 2. Please give the technical justification for using "0.1 percent gap activity" in the subsequent analyses.