APR 20 1983

Docket No. 50-225

Dr. Donald R. Harris, Director
Critical Facility
Department of Nuclear Engineering
and Science
Rensselaer Polytechnic Institute
Troy, New York 12181

Dear Dr. Harris:

Our contractor, Los Alamos National Laboratory, has identified a need for additional information to supplement the information you provided in your amended license renewal application submittal.

Please refer to the enclosed list of questions and have the responses available by May 23, 1983, when the LANL and NRC staff are planning a site visit to Rensselaer to discuss the various aspects of your reports and the NRC Safety Evaluation Review.

If you have any questions, please contact your Project Manager, Hal Bernard at (301) 492-9799.

Sincerely,

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
As stated

cc: J. Hyder, LANL

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RENSSELAER POLYTECHNIC INSTITUTE
PRELIMINARY QUESTIONS

1. Your SAR (submitted January 1983) analyzes a rod withdrawal accident that inserts a ramp reactivity worth of .32% $\Delta K/K$, the total excess reactivity available in your current core configuration. Although this calculation represents the most severe reactivity transient for your current core, it does not represent the worst case that can be postulated based on your Technical Specifications, which allow a maximum excess reactivity of 3.9% $\Delta K/K$. Because the ^{235}U inventory at your facility has been reduced recently to <5.0 kg, you should reevaluate your current requirements on excess reactivity and, if appropriate, propose a Technical Specifications change. In any case, the reactivity transient accident should be reanalyzed to conform with your Technical Specifications limit.

2. In addition to the reactivity accident, consider and address potential accidents caused by natural hazards such as earthquake, fire, windstorm, and flooding, and a potential mechanical rearrangement of the fuel caused by an externally initiated event.

3. Comment on the ability of the reactor components and systems to continue to operate safely and withstand prolonged use over the term of the requested license renewal. Include the potential effects of aging on fuel elements, instrumentation, and safety systems.

4. What are the reactivity worths of the control rods in your current core configuration? Which absorbers are currently in use?

5. Provide a drawing or schematic of the control rod drives. Describe in detail the position indications available at the control console including both readouts and lights.

6. Provide further description of the reactor tank including wall thickness, dimensions of lower extension, and location and size of all penetrations. Is the tank completely stainless steel as stated on p. 22 of the SAR, or is it stainless-steel lined as stated in Technical Specification 5.4.1?

7. Provide further detailed information on the water handling system. What are the dimensions of the dump, fill, and drain lines? Note that your current SAR (p. 38) refers to a 5-in.-diam dump line, whereas the FSAR submitted in September 1979 refers to a 6-in.-diam line. Describe the fail-safe gate valve in the dump line. Is it electrically or pneumatically actuated? What are the maximum reactivity insertion rates at the slow and fast pump speeds? Is there a water level sensor in the reactor tank and a level indicator at the console? Describe the water level interlock specified in Table II of the Technical Specifications.

8. Provide more information on the reactor support system including the thickness of the three grid plates, dimension of the support posts, mechanisms of force transmittal from grid plates to support posts, and attachment of posts to the reactor tank.

9. How many manual scram buttons are there and where are they located? Where is the "manual electrical switch" located?

10. The Technical Specifications limit the maximum allowable reactivity insertion rates to 0.084% $\Delta K/K/s$ at low multiplication and 0.033% $\Delta K/K/s$ at high multiplications and critical conditions. Describe how these maximum rates are assured for both water addition and control rod movement. If either mechanical or electrical means are used, what would the maximum insertion rate be in the case of a failure in the limiting means, such as a throttling valve inadvertantly being driven fully open?

11. Can more than one control rod be moved at the same time? If so, how many can be moved simultaneously?

12. Do interlocks and/or procedures allow attaining criticality during water addition; that is, can control rods be removed and water then be added?

13. Do your procedures allow water to remain in the tank when the reactor is secured or unattended?

14. Is the Ag-Cd-In control absorber clad in stainless steel? (Note that Technical Specification 1.L.1 says no, and the SAR p. 27 states yes.

15. How was the moderator temperature coefficient measured? In particular, how were the moderator and fuel contributions separated?

16. SAR, p. 42 - provide reference for "Collier correlation."

17. SAR, p. 45 states the less negative value of fuel temperature coefficients reported in Reference 3 was used in your analysis. Table 14.4, p. 55 shows the average value. Which was actually used in the calculation?

18. SAR, p. 45 - Further explain and/or derive equations 7.4 and 7.5.

19. SAR, p. 60 - Column headed "Moderator Temperature Coefficient" is confusing. Neither units or value seem to be correct. Coefficient is function of temperature, not a constant value as shown. Could values be reactivity changes as opposed to temperature coefficients?

20. SAR, p. 65, Fig. 14.3 - The assembly width of 2.96 in. is not compatible with same dimension shown as 7.2644 cm. The pitch dimension shown near the bottom of the figure as .015/in. cannot be correct. The thickness of the side plate shown as .020 is in disagreement with SAR description (p. 25) of .027 in. Make appropriate corrections.

21. Technical Specification 5.4.2 should be revised to reflect the current conditions.

22. Describe the administrative organization of the radiation protection program, including the authority and responsibility of each position identified.

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

24. Describe any radiation protection training for the non-Health Physics staff. If possible, provide a topic outline of the courses and indicate the normal duration of each course or lecture.

25. Summarize your general radiation safety procedures. Identify the minimum frequency of surveys, action points (levels), and appropriate responses.

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

27. For the fixed-position radiation and effluent monitors, specify the generic types of detectors and their efficiencies and operable ranges.

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

29. Identify the generic type, number, and operable range of each of the portable Health Physics instruments routinely available at the critical facility. Specify the methods and frequency of calibration.

30. If you anticipate that additional or specialized Health Physics instrumentation may be needed from other nearby University facilities, indicate the type, number, and range of the available equipment.

31. Describe your personnel monitoring program. Describe calibration procedures for any in-house portions of this personnel monitoring program. Describe any Quality Assurance studies for the commercially supplied portions.

32. Identify any administrative personnel exposure limits and 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.

33. Provide a summary of the critical facility's annual personnel exposures [the number of persons receiving a total annual exposure within the designated exposure ranges, similar to the report described in 10 CFR 20.407(b)] for the last 5 yr of operation.

34. List all parameters that are alarmed in the control room and specify alarm trip settings.

35. Describe your environmental monitoring program; summarize the results for past 10 yr and compare recent measurements with any performed before the initial criticality.

36. Provide a description of the coolant system, include demineralizer system, pumps, valving, and other important features.

37. Provide a description of the electric power system, include bus description, phase and cycle information, and facility distribution system.

38. List all control, lighting, or instrumentation that is provided with emergency back-up power.

39. Describe the technique used to calibrate power levels.

40. What is the minimum leak rate (expressed in gallons per minute) that can be readily detected in the coolant system?

41. Have all modifications and proposed modifications to the instrumentation and control system been formally documented?

42. Describe the auxiliary systems such as compressed air, cranes, and so on.

43. Describe the fire protection system. Specify who is responsible for the maintenence of facility fire extinguishers.

44. Identify all instrumentation that is on a regular calibration cycle.

45. Describe the facility ventilation system, include locations of intakes and exhausts, dampers, filters, and flow rates.

46. List and describe specifically any engineered safety features. ("Engineered Safety Features" are those provided to mitigate the consequences of postulated accident, such as automatic shutdown of ventilation system).

47. Is the city water system protected from back-flowing of water from the coolant storage tank?

48. Specify all nuclear and process parameters that are monitored by redundant instrumentation.