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Docket No. 50-116

September 28, 1983

Mr. Cecil O. Thomas
Chief, Standardization & Special Projects Branch
Division of Licensing, NRR
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Thomas:

Enclosed are the responses to the Formal Questions sent to Iowa State University under your cover letter dated August 15, 1983. In a telephone conversation with the Project Manager, Ms. Chu, an extension of the submittal deadline to September 30, 1983, was arranged.

If you have any questions, please contact me at (515) 294-6422 or 294-5840

Sincerely,

RAHendrickson

Richard A. Hendrickson
Reactor Manager

Enclosure: As cited

cc: B. I. Spinrad, Chm, Nuc E Dept
L. E. Burkhart, Chm, Univ Rad Safety Com

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RESPONSES TO FORMAL QUESTIONS

Issued August 15, 1983

1. Q: The SAR (p. 4-48) states that 476 full-power hours "exceeds the historical highest annual usage of the UTR-10 by a factor greater than four." Table 4.2-3 shows usage for the 1965 calender (sic) year of 510 h full-power. Presumably you meant "full-power annual usage." Please clarify.

   A: The question applies to page 4-38, not 4-48, of the SAR. The phrase "... estimate of the full-power hours required to reach the average annual concentration limit for unrestricted areas is 476 hours." was intended to be equivalent to 4760 kilowatt hours per year, since the full-power rating on the reactor is 10 kW. This is more than the highest-usage year (1968) by a factor greater than four, when 1000 kWh were recorded.

2. Q: Are all scrams full scrams as defined by Technical Specifications (TS) Sec. 1.19? If all are full scrams, you might consider revising the rod-drop scram definition in the TS.

   A: All scrams, if scram implies an automatic response to a predetermined safety trip setting, are full scrams. Reference to rod-drop scrams has been removed from the revised Technical Specifications (August 1983).

3. Q: Identify the generic type, number, and operable range of each of the portable Health Physics instruments routinely available at the reactor installation. Specify the methods and frequency of calibration.

   A: Refer to Attachment #1.

4. Q: Describe the facility's fire protection system.

   A: Fire/smoke detection is based on personal observation; manually activated evacuation alarm switches are located in first-floor hallways on opposite sides of the Laboratory. The emergency phone number to the Ames Fire Department is attached to each telephone cradle. Fire extinguishers are kept in the reactor room -- 1 class ABC and 1 class C -- and in the area immediately to the south of the reactor room -- 1 class A.

5. Q: The Iowa State University UTR-10 had a number of modifications to reactor systems during the period from 1960 to present. Update the list of modifications in Section 1.3 of the SAR, describe the reasons for the modifications, e.g., increased operations flexibility, correction of design or operational deficiencies, etc., and address the safety significance of the modifications.

   A: Refer to Attachment #2.

6. Q: In the SAR Section 2.3, you have stated the frequency of winds in excess of 38 miles per hour is on the order of 0.001. Clarify the unit of the frequency, e.g., per month or per year.

   A: The unit of frequency, as shown in the footnote to Table 2-4 (incorrectly cited in paragraph 2.3 (b) on p. 2-3 as Table 2-3), is per 5 years.

7. Q: Amplify the information in your SAR 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.

   A: Prediction of lifetime of components and systems without failure, whether due to aging or other reasons, is bare speculation without an adequate statistical data base relating to failure frequencies. That data base is not available. Rather than addressing the issue of withstanding prolonged use, attention should be focused on detection of deterioration of performance of components and systems important to the safe operation of the reactor facility.

   In general, all safety-related components and systems are tested as a part of the surveillance program required by technical specifications. If properly designed and interpreted, these tests provide an adequate source of information on the condition and ability of these components and systems to perform their intended safety functions.

   Specifically, as fuel elements age, the principal concern is leakage of fission products as corrosion of the cladding progresses; no other safety-related effects due to aging are credible, and surveillance activities are sufficient to detect cladding failure. Instrumentation important to safety is periodically calibrated and tested. The chief concern in this area is the subtle failure that reduces performance; however, the proper application of the limiting conditions for operation and the operating procedures minimize the chance of a significant safety-related failure. The design and construction of the instrumentation system utilizes, for the most part, electronic components which are commercially available with little delay in procurement; a modest critical spare parts inventory provides those parts which are scarce. The performance of safety system components is tested periodically during prescribed surveillance activities, and the main concern is the availability of sufficient negative reactivity to safely shut down the reactor when needed. The control rods and their drive mechanisms reveal aging effects in withdrawal and insertion timing and reactivity worth measurements required in the technical specifications; the dump valve opening time is also measured. Withdrawal, insertion and opening times are compared with previous data to detect trends which would indicate deterioration of performance.

8. Q: Describe the administrative organization of the radiation protection program, including the authority and responsibility of each postion identified.

A: Refer to Attachment #3.

Answer to Question #3 of letter from Nuclear Regulatory Commission to Dr. R.A. Hendrickson dated August 15, 1983.

The generic type, number, and operable range of each of the portable Health Physics instruments routinely available at the reactor installation are listed below:

Type	Number	Operable Range
Direct reading pocket ionization chamber	39	0-200 mR
"	4	0-1 R
"	9	0-20 R
"	17	0-200 R
Ionization chamber survey meter	1	0-5 R/hr
"	1	0-300 R/hr
"	1	0-10K R/hr
Geiger-Mueller survey meter	3	0-50K cpm
Proportional alpha counter survey meter	2	0-500K cpm
Neutron survey meter	1	0-2 Rem/hr
"	1	2-4K $n/cm^2/sec$

All portable monitoring equipment is calibrated quarterly, except for direct reading pocket ionization chambers which are calibrated and tested for leakage on an annual basis. At the time of survey meter calibration, batteries are removed and checked under load. All gamma calibrations are done using a J.L. Shepherd shielded calibration range equipped with a dual ^{137}Cs source calibrator and a system of attenuators. Exposure rates at given distances for this calibrator were obtained using instrumentation whose calibration is directly traceable to the National Bureau of Standards.

Geiger-Mueller survey meters are checked electronically on each scale with a known input pulse rate. Each scale is then calibrated at two points to within ten percent accuracy using the shielded calibration range described above.

Ionization chamber survey meters are calibrated at two points on each scale to within ten percent accuracy using the shielded calibration range described above.

Proportional alpha counter survey meters are checked for proper gas flow and calibrated at six points on the linear - log scale to within ten percent accuracy using an Eberline Instruments Corporation certified plutonium calibration set.

Neutron survey meters are calibrated using a standard one curie plutonium-beryllium source with emission rate of 1.64×10^6 n/sec. These instruments are calibrated at two points on the x1 and x10 scales and one point on the x100 scale. Scales not calibrated are noted on a sticker affixed to the instrument.

Sep 1973	Replaced shim rod position potentiometer mounting hardware with improved version.
Feb 1974	Added circuit to one of two period trip test circuits and wiring to down-limit switches to provide start and stop signals for rod drop timing measurements.
Jul 1974	Added a HEPA filter to the rabbit tube exhaust.
Aug 1975	Amendment #5 to license. Lowered limit of contained U-235 as fuel elements to 4.6 kilograms reflecting return of loaned fuel used as an alternate core.
Oct 1976	Installed a standby AC power supply system for the radiation monitoring channels and alarm.
Jan 1978	Added a time-delay relay to the scram circuit to provide a minimum delay of 3 seconds before the scram circuit can be reset following a scram.
Mar 1978	Installed two new area radiation monitoring channels (GA) and one equipment radiation monitoring channel (GA) as equivalent-function replacements of old equipment.
May 1978	Added a relay logic system to the radiation monitoring alarm system to provide selection of trip level before evacuation signal occurs depending on reactor operation status.
Jul 1978	Modified radiation alarm system logic power supply to provide relief from false alarms during switchover to standby power during AC power failure.
Nov 1978	Added wires to up-limit switches to provide stop signal for scram circuit delay time measurements.
Apr 1981	Installation of buffer amplifiers for use of safety-related electronic signals in various measurements and experiments.
Mar 1982	Connection of a dual-channel rod-drop timing gate circuit to measure delay and rod-drop times using only one rod drop.
May 1983	Process instrumentation system installation of electronic calibration/test circuit for calibration and testing of process instruments.

MODIFICATION DATE	REASON FOR MODIFICATION	SAFETY SIGNIFICANCE
4-60	Requested by manufacturer	Fuel is forced into a position of Maximum Reactivity
1-63	Low Power Core for experiments that required a reduced gamma background	More frequent fuel transfers, increases radiation exposures
4-63	Accurate positioning of experiments in shield tank	None
	Fewer changes of primary ion resin	Less exposure to radioactivity in primary resin plus lower conductivity water being added to primary from shield tank
	Requested by AEC	Requires constant operator attention
10-64	To reduce moisture induced electronic noise	None
3-65	Protects the fuel and reduces the introduction of foreign material into the coolant	Reduces radiation levels due to less activation of foreign material
7-71	Increased reliability, decreased repairs and spurious period scrams	Improved reliability on safety system testing
9-71	Individual calibration of each rod position could be performed without affecting the other rods position indication	None
9-73	To correct potential misalignment of position pot	Reduce probability of jamming shim safety during rod insertion
2-74	Permit accurate timing measurements	Period trip will work normally when circuit is disconnected
7-74	Requested by NRC	Reduces the chance of accidentally spreading radioactive material
8-75	Required by NRC to reduce holdings below a strategic mass	Reduced radiation levels during fuel transfer
10-76	Provide constant power to the area monitor system during power outages	Earlier detection of radiation fields

MODIFICATION DATE	REASON FOR MODIFICATION	SAFETY SIGNIFICANCE
1-78	Requested by NRC	All water is removed from core tanks prior to restart
3-78	Replacement of old equipment that was too expensive to repair.	More accurate and reliable
5-78	Raise limit on evacuation horn while operating reactor	None
7-78	Decrease false alarms during power outages	Decrease false alarms during power outages
11-78	Needed to know delay time values	Surveillance of safety system response
4-81	Use safety related electronic signals in various measurement and laboratory experiments	No safety-related instruments will be affected
4-82	Savings in maintenance during the expected service life of rods and in surveillance time	Fewer drops of the rods resulting in less wear on equipment
5-83	Decrease time spent on surveillance activities related to calibration	Allow testing of trip setpoints in process instruments during precritical checks

Answer to Question #8 of letter from Nuclear Regulatory Commission to Dr. R.A. Hendrickson dated August 15, 1983.

General review and audit of radiation safety programs at Iowa State University is provided by the Radiation Safety Committee, appointed by the Vice President for Academic Affairs. This committee has the authority to require program changes, including termination of unsafe projects. The Radiation Safety Committee determines policies concerning the use of ionizing radiation and must give approval before ionizing radiation is used in any specific project.

The Radiation Safety Officer is a member of the Radiation Safety Committee and is normally charged with carrying out the directives of the Radiation Safety Committee. The Radiation Safety Officer operates out of the Department of Environmental Health and Safety which has broad responsibilities for all safety and occupational health programs on the Iowa State University campus.

The Reactor Use Committee is appointed by the Radiation Safety Committee and reports to the Radiation Safety Committee. The Reactor Use Committee reviews new experiments and proposed alterations to the UTR-10 reactor and reviews and audits reactor operations for safety and regulatory compliance.

The Radiation Safety section of the Department of Environmental Health and Safety provides consultation service on radiation problems; dosimetry services; ionizing radiation project review; radiation and contamination surveys; accident response; radioactive waste collection and disposal; safety inspections for regulatory compliance; and other radiation related services. This is all accomplished under the direction of the Radiation Safety Officer.