GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS OPERATIONS
GENERAL ELECTRIC COMPANY • VALLECITOS NUCLEAR CENTER • PLEASANTON, CALIFORNIA 94566

June 19, 1984

Cecil O. Thomas, Chief
Standardization and Special Projects Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: 1) License R-33, Docket 50-73.
2) Application for Renewal of License R-33; June 13, 1979.
3) Letter, C. O. Thomas to R. W. Darmitzel; June 1, 1984.

Dear Mr. Thomas:

Enclosed are our responses to the questions concerning the renewal of the Nuclear Test Reactor (NTR) license contained in your letter of June 1, 1984 (Ref. 3).

Revised proposed Technical Specifications for the NTR also will be forwarded to you by July 6, 1984.

Sincerely,

D.E. Cunningham

G. E. Cunningham
Senior Licensing Engineer
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Certified By S. Anderson

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GENERAL ELECTRIC

AFFIRMATION

Nuclear Test Reactor License Renewal Information

To the best of my knowledge and belief, the information contained in the enclosed document is accurate.

By______________________________
R. W. Darmitzel, Manager
Irradiation Processing Operation

Submitted and sworn before me this _________ day of June, 1984.

________________________, Notary Public in and for the County of Alameda, State of California.

GENERAL ELECTRIC NUCLEAR TEST REACTOR
Response to Review Questions

1. Question

Provide Technical Specification for an acceptable conductivity range in the primary water and include appropriate measurement and calibration frequencies.

Answer

Proposed Technical Specification 3.3.3.2 has been added and Table 4-2 has been revised to include conductivity limits, surveillance and instrument calibration.

2. Question

What is the average weight in grams of the ^{235}U in the fuel disk for the current core?

Answer

Unirradiated new fuel in 1957 = 3,992 gm U in core
@ 93.17% enriched = 3,719 gm U-235
with 640 Discs = 5.81 gm U-235/disc (average)

At Jan. 1, 1984, the NTR has operated 90 MwD

$$90 \text{ MwD} \times 1.27 \frac{\text{gm U-235 burned}}{\text{MwD}} = 114 \text{ gm U-235 burned (total for 640 discs)}$$

$$\frac{114 \text{ gm U-235 burned}}{640 \text{ discs}} = 0.179 \text{ gm U-235 burned/disc (average)}$$

At Jan. 1, 1984

U-235 per disc = 5.81-0.18 = 5.63 gm (average)

3. Question

Define potential, console, and total excess reactivity. How are they determined? Examples would be very helpful.

Answer

Potential excess reactivity is defined in proposed Technical Specification 1.15 as that excess reactivity which can be added by the remote control of poison rods plus the maximum credible reactivity addition from primary coolant temperature change plus the potential

reactivity worth of all installed experiments. (See proposed Technical Specification 1.16.)

Potential excess reactivity is determined by utilizing rod calibration curves for remotely movable poison rods, a primary coolant temperature coefficient of reactivity curve and calculated or measured reactivity worths for all installed experiments.

Console excess reactivity as used in NEDO-12727, Table 4-1, is a misnomer for the value (K_{eff}-1) given in units of $\Delta K/K$ and \$ for the arrangement of movable poisons specified in the typical operational core which had 0.3 \$ potential excess reactivity.

"Console excess reactivity" (actually K_{eff}-1) is determined for other movable poison configurations by adding or subtracting the following reactivity worths:

3 control rods worth	=	2.3 \$
4 safety rods worth	=	2.0 \$
2 MPS (Manual Poison Sheets) worth	=	0.9 \$

The values given in Table 4-1 were intended to provide general information about reactivity effects. The term "console excess reactivity" is obsolete and no longer used. The term is not used in the proposed Technical Specifications.

Total excess reactivity is the excess reactivity with the reactor in the most reactive condition. Elsewhere it is also called excess reactivity, built-in reactivity or available reactivity. For the NTR it is the potential excess reactivity plus the reactivity worth that would be added by removing all manual poison sheets (MPS) and making the reflector complete. The total excess reactivity is determined below using the reactivity values currently associated with the reactor setup for neutrography.

Potential excess reactivity	=	.55\$
3/8 MPS in Position #1	=	1.06
3/8 MPS in Position #5	=	.91
Graphite exchanged for Source Log	=	.50
Total excess reactivity		3.02\$

Total excess reactivity is determined by adding previously measured reactivity values. Total excess reactivity is a hypothetical value for a reactor condition that will never exist. Technical Specifications do not permit reactor operation in this condition.

4. <u>Question</u>

What is the shutdown margin? How is it determined? Provide a Technical Specification for shutdown margin. Provide calculations with highest worth rod out.

Answer

The shutdown margin is that amount of reactivity by which the reactor would be subcritical if all control rods were fully withdrawn from the reactor and the strongest safety rod removed (assumed stuck out).

Shutdown margin is determined by calculation as follows:

Total worth for all four safety rods	3.86$
Less maximum permitted potential excess reactivity (control rods, temperature, experiments)	-.76$
	3.10
Less strongest safety rod	-1.10
Shutdown margin (with the strongest safety rod removed)	2.00

New Technical Specifications are provided related to shutdown margin. See Proposed Technical Specifications 1.24 (revised), 3.1.3.3 and 4.1.3.3.

5. Question

What are the accuracy and precision (reproducibility) for each control rod position measurement?

The accuracy of the control rod position readout is 0.01-inch. Comparison of the position indication and the actual control rod position (measured) is performed periodically and the difference is required to be less than± 1-inch at the full-out position of 16-inches for the coarse rods and ± ½-inch at the full-out position of 15-inches for the fine rod.

6. Question

What are the individual worths of each safety and control rod and each manual poison sheet? How are these worths determined?

Answers

Control Rods:

CR #1	.59$
CR #2	1.01$
FR	.60$

These were determined by measuring reactor period for differential rod movements.

Safety Rods:

SR #1	.82$
SR #2	1.10$
SR #3	1.07$
SR $4	.87$

These were determined by measuring the reactor critical rod position and then utilizing control rod calibration curves to calculate reactivity worths.

Manual Poison Sheets:

3/8 MPS in Position #5	1.06$
3/8 MPS in Position #1	0.91$
3/4 MPS in Position #1	1.37$

These are based on critical rod position measurements.

7. Question

How many of the poison sheet positions have been modified to provide positive restraint for the manual poison sheet? If less than six, what controls exist to insure that only the modified positions are used? Explain all normal operations involving the manual poison sheets.

<u>Answer</u>

Three Manual Poison Sheet (MPS) positions (#1, 2 and 5) have been modified to provide positive restraint for the manual poison sheets. All Manual Poison Sheet movements are strictly controlled administratively in accordance with approved procedures.

The reactor is currently operated in two configurations. In the neutron radiography mode, a 3/8 MPS is in position #1 and a 3/8 MPS is in position #5. In the reactivity testing mode a 3/4 MPS is in position #1 and position #5 retains a 3/8 MPS. When changing modes (while the reactor is shut down), a full MPS is inserted into position #2, the MPS changed in position #1 and the full MPS removed from position #2. The two operating modes and the changeover are the only normal operations involving the Manual Poison Sheets.

8. <u>Question</u>

What is the source of supply air to the reactor facility ventilation air?

Answer

Supply air to the reactor cell is atmospheric air from the areas adjacent to the cell. Air is drawn through the various wall penetrations including the thermal column penetration between the reactor cell and the south cell. Air flow is accomplished by the exhaust fan which provides a slightly negative pressure in the reactor cell.

9. Question

How do you ensure that the primary coolant level in the fuel loading tank is maintained at the desired level?

Answer

Water level in the fuel loading tank is checked and the tank is manually filled monthly. This frequency is adequate to assure a proper level in the tank. The tank is provided with a float switch which activates a visual and audible alarm in the control room if an unusual circumstance were to cause the water level to drop below or raise above predetermined levels between monthly checks.

10. Question

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

Answer

Radiation protection services are provided at the NTR by the Nuclear Safety group. These services include radiological engineering, criticality safety, training, emergency coordination, radiation monitoring and compliance engineering.

a. Radiological Engineering

Provides technical standards for the safe and efficient operation of nuclear activities, radiological criteria for acquisition, construction and modification of facilities and equipment; and technical safety support.

b. Criticality Safety

Provides authoritative professional advice and counsel to managers and supervisors on matters of control against accidental criticality and measures the effectiveness of the criticality control program. Conducts analyses and reviews operations to ensure they conform to the physical situations on which the calculations of criticality limits have been based.

c. Site Training and Emergency Coordination

Prepares and administers courses and other evaluation techniques in the areas of nuclear safety to assist operating management in achieving and maintaining the level of competence necessary to carry out an effective safety program. Develops and maintains an emergency plan and coordinates with operating management in developing implementing procedures for coping with emergencies.

d. Radiation Monitoring

Assists facility personnel in the control of contamination, exposure to individuals, monitoring for disposal of wastes, and the reliability of radiation detection instruments including appropriate records and reports associated therewith.

e. Compliance Engineering

Provides review of plant operations and related activities and of proposed tests, experiments, operating procedures and facility modifications.

Radiation protection program personnel have all necessary authority to perform the assigned responsibilities.

11. Question

Describe any radiation protection training for the non-Health Physics staff.

Answer

All new employees receive an Initial Radiological Safety Orientation course (20 min.) as part of their sign-in exercise. Within 30 days of hire, all new employees receive a New Employee Radiological Safety Orientation (4 hour). Employees take a Respiratory Protection Training course if appropriate.

All NTR operations personnel become licensed operators pursuant to 10CFR, Part 55. As licensed operators, each person participates in an Operator Requalification Program which requires the individual to maintain and demonstrate proficiency in radiation protection.

12. Question

Provide information on GE-NTR "ALARA" Policy.

Answer

Maintaining radiation exposure as low as reasonably achievable (ALARA) is an established policy at NTR. Facility and equipment design and modification and new experiments consider radiation exposure reduction when appropriate. Periodic reminders to employees of the importance of maintaining exposures ALARA are conducted through letters or safety

meetings. There is an independent ALARA review conducted by Compliance Engineering.

13. Question

For the fixed-position and effluent monitors, specify the generic type of detectors and their operable ranges. Also, describe the frequency and methods of instrument calibration and routine operational checks.

Answer

a. Type and range

Description of Monitor	Generic Type	Operable Range
Stack Particulate	Thin-window G.M.	10 to 1×10^{7} cpm
Stack Noble Gas	Flow-through ion chamber (Kanne)	10^{-13} to 10^{-7} amps (10^{-6} to 1×10^{0} μCi/cc)
Remote Area Monitors	Ion chambers	0.1 to 10^{7} mR/hr.
Reactor Cell CAM (Air Monitor)	Thin-window G.M.	50-50K cpm
Hand-and-Shoe Monitor	G.M. Tubes	0-500 cpm

b. Calibration method and frequency.

Description of Monitor	Calibration Method	Frequency
Stack Particulate	1. Count rate meter set with electronic pulser.	Quarterly
	2. Detector efficiency checked with Cl-36 standard	Quarterly
	3. Sample flow controller set with calibrated rotameter	Semiannually
Stack Noble Gas	1. Calibrate response with known concentration of Xe-133, and source check with Co-60 V-block source externally.	Initially
	2. Test response with same Co-60 V-block source.	Quarterly
Remote Area Monitors	1. Perform gamma calibration (Co-60) and linearity check	Initially or After Repair

	2. Response test to internal Cl-36 source.	Quarterly
Reactor Cell CAM	1. Measure counting efficiency with Cl-36 standard.	Quarterly
	2. Measure flow rate and adjust flow controller.	Semiannually
Hand-and-Shoe Monitor	Source checks by Instrument Technician (pulse check, clean, source check, verify alarm settings).	Semiannually

c. Routine operational checks

1) Stack particulate and gas monitors

- Check and record indication prior to initial startup each day.

- Record indication after reaching stable reactor power and approximately every hour for the remainder of the run.

- Check the HI GAS ACTIVITY ALARM operation and setpoint monthly.

2) Remote area monitors

- Record the indication and trip setpoint for the reactor cell, south cell, and cleanup system demineralizer channels prior to initial startup each day.

- Record the indication on the reactor cell, control room and north room after reaching stable power and approximately every 2 hours for the remainder of the run.

- Check the HIGH RADIATION alarms and the calibration with the internal check source for the north room, reactor cell, control room, MSM, and south cell each month.

3) Reactor cell CAM

- Check operation and indication prior to initial reactor cell entry each day and observe indication prior to each cell entry.

- Check the efficiency with a Cl-36 source weekly.

4) Hand-and-Shoe Monitor

- Daily source check on function and alarm points.

- Monthly survey at the hand-and-shoe counter location, record radiation background levels.

14. Question

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.

Answer

Generic Type	Number	Operable Range	Calibration Method*	Calibration Frequency
Ion Chamber, Dose Rate:	2	0 to 250 R/hr	1	Initially, After Servicing, and Annually
	2	1 mR/hr to 1,000 R/hr	1	"
G.M., Dose Rate	2	0 mR/hr to 1,000 R/hr	1	
G.M., Count Rate	6	0 to 60,000 cpm	2	"
Alpha, Scintillation	3	0 to 2,000,000 cpm	3	"
Neutron Dose Rate	1	0 to 5,000 mRem/hr	4	"

*Calibration Methods

1. Multipoint calibration in known Co-60 gamma field.
2. Electronic pulse rate input and response test with beta source.
3. Probe placed in contact with four-decade range of Pu-239 standards. Output adjust to approx. 16% efficiency.
4. Electronic pulse input and AmBe neutron source response check.

The portable health physics instruments listed above are routinely available at the NTR facility. Other instruments at other on-site laboratories and facilities are available for NTR use should the need arise.

15. Question

Describe your personnel monitoring program.

Answer

a. External Dosimetry

NTR personnel wear beta-gamma film which is changed and read monthly and neutron albedo dosimeters which are changed and read quarterly. Self-reading gamma pocket dosimeters are used to estimate the penetrating whole body exposure between times of monthly badge processing. For high dose rate work, TLD finger rings, alarming dosimeters and timekeepers may be utilized. One person is assigned responsibility for maintaining exposure records and assuring all personnel are notified of their exposure.

b. Bioassay

The bioassay program provides for both in vivo counting and measurement of radioactivity in urine, feces, breath and other biologically derived samples. In vivo counting is accomplished primarily with the aid of a 5-inch by 5-inch NaI scintillation detector and a multichannel analyzer.

The frequencies and types of measurements depends on the work being performed. At the present time, NTR operations personnel receive in vivo counting quarterly and a gross beta measurement of a urine sample annually.

16. Question

Provide a summary of the reactor 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 years of operation.

Whole Body Exposure Range (rems)	Number of Individuals in each range by year				
	1979	1980	1981	1982	1983
No measurable exposure	0	0	0	1	1
Measurable <0.1	0	0	2	1	0
0.1 to 0.25	0	0	1	1	1
0.25 to 0.5	3	0	0	0	0
0.5 to 0.75	1	0	3	0	0
0.75 to 1	1	1	0	0	0
1 to 2	2	4	3	2	5
2 to 3	0	0	0	3	0
>3	0	0	0	0	0

17. Question

Specify the concentration (μCi/mL) of noble gases that will equal the stack alarm point of 2×10^{-11}. A. Specify the quantity (μCi) or the concentration (μCi/mL) of airborne particulates that will produce the stack alarm point of 1×10^4 cpm.

Answer

See NEDO 12727, Appendix A, p. 15.

Noble Gas Alarm Point $= 2 \times 10^{-11}$ amps $= \underline{2 \times 10^{-4}\ \mu Ci/cc}$

Particulate Alarm Point $= 1 \times 10^4$ cpm

$$\frac{10{,}000 \text{ cpm}}{0.04 \text{ c/d}} \times \frac{\mu Ci}{2.22 \times 10^6 \text{ dpm}} = 0.113\ \mu Ci \text{ collected on the filter}$$
(assuming 0 cpm background)

at a sample flow rate of 1 cfm and a stack flow rate of 3,000 cfm.

This is equivalent to a total β,γ particulate release of:

$3{,}000 \times 0.113\ \mu Ci = \underline{340\ \mu Ci}$

from the stack during the sample collection period.

18. Question

What is the average annual release of ^{41}Ar from the reactor facility to the environment? Provide data for the last 5 years.

Answer

The noble gas stack effluent monitor provides a measurement of the gross radioactivity from the mixture of noble gas released. The stack limits are based on worst case isotope and isotopic records are not maintained. A measurement in approx. 1969 showed that the Ar-41 release rate during 100 kw operation was 55 μCi/sec. Based on this release rate and the reactor power history estimated Ar-41 release would be as follows:

Year	Ar-41 Release (Ci)
1979	175
1980	204
1981	173
1982	197
1983	126

19. Question

Describe the liquid radwaste management program.

Answer

Contaminated "waste" water either drains directly to the 500-gallon holdup tank located in the reactor cell or drains to the approx. 50-gallon reactor cell sump and then is pumped into the holdup tank.

Sources piped directly to the holdup tank are:

1. fuel loading tank overflow
2. primary coolant system air trap vent
3. primary heat exchanger drain line

Sources going to the reactor cell sump are:

1. primary sample station sink
2. pump seal leakage
3. floor drainage

There has been no significant amount of water accumulated in the holdup tank since 1976. Daily venting of the primary system is the major contributor, and the amount of water "vented" from the primary system is not measured, but it is small. Total makeup to the primary system is less than 20 gallons per month. The water collected by the sump is also small and normally evaporates before being pumped to the holdup tank.

If it should be necessary or desirable to get rid of liquid radwaste, it would be transferred from the holdup tank to the VNC Waste Evaporator facility which is licensed by the State of California and the NRC. Prior to the transfer, the water would be sampled. The VNC chemistry laboratory would analyze the sample for:

1. pH
2. alpha concentration
3. beta concentration
4. total uranium
5. uranium enrichment
6. gamma scan for radionuclide identification
7. tritium concentration

There has been no liquid radwaste released or shipped from the NTR facility since 1976.

20. Question

Describe the solid radwaste management program.

Answer

Solid radwaste at NTR is collected in bags situated at the "step-off pad" at the entrance to controlled areas where contamination is present. Solid radwaste may also be transferred directly to steel drums or specially fabricated wood boxes. Solid radwaste is stored at the facility in locked interim storage areas.

Periodically the radwaste is transferred to a site wide radwaste storage area. At these locations another group is responsible for contents inspection and final preparation for shipment in conformance with all applicable regulations.

The solid radioactive waste shipped from the NTR for the last three years is as follows:

1982 - Solid noncompactible waste 22.5 ft^3, 0.406 Ci
- Hardware containing SNM 2.75 $ft.^3$, 0.25 Ci
- Uncompacted compactible waste 108 $ft.^3$, 0.0007 Ci

1983 - Solid noncompactible waste 256 $ft.^3$, 0.221 Ci
- Uncompacted compactible waste 108 $ft.^3$, 0.0006 Ci

1984 - None to date

Prior to 1982 the NTR facility was part of the same organization as the

General Electric Test Reactor (GETR) facility. Radwaste records of both facilities were combined and records did not identify the origin of the waste.

21. Question

As part of 19 & 20, include summaries of the quantities of liquid and solid radioactive waste resulting from reactor operation for the last 5 year (total activity of each physical form at times of release or shipment for each year).

Answer

As part of questions 19 and 20 these summaries were included.

22. Question

Describe your environmental monitoring program; summarize the results for the past 5 years and compare recent measurements with any performed before any initial reactor criticality.

Answer

In addition to the sampling and monitoring of liquid and gaseous effluents, VNC has established a program of environmental measurements both on and offsite to assure that there is no reconcentration of radioactive materials resulting from all site (not only NTR) activities. Wells are sampled both on and off site. Soil, stream bottom and vegetation samples are taken, both-on and off-site. Four air sampling stations are positioned approximately 90° apart around the operating facilities of the site. Membrane filters from each station are changed and counted weekly. Perimeter stations are located on site to measure cloud gamma radiation.

Data from the environmental monitoring program for the past 5 years is summarized in Tables I-VI. Table VII presents a summary of data collected in 1957 which is the second year of site activities.

Summary of Environmental Results
1979 - 1983

Table 1: Average Annual Radioactivity Concentrations in Receiving Waters (pCi/l)

Year	α	β,γ	Tritium
1979	<7.82	<27.55	
1980	<1.51	<5.23	$\leq 1.90\times10^3$
1981	$\leq$0.57	$\leq$4.78	$\leq 0.94\times10^3$
1982	$\leq$0.69	$\leq$5.19	$\leq 1.75\times10^3$
1983	$\leq$0.48	$\leq$4.56	$\leq 2.53\times10^2$

Table II: Average Annual Radioactivity Concentration in Stream Bottoms ($\mu Ci/gm$)

Year	α	β,γ	Co-60	Cs-137	Sr-90	Pu-239
1979	≤7.57	14.15	≤0.805	≤2.83	≤0.162	≤0.020
1980	≤0.734	10.80	≤0.388	≤5.23	≤0.072	No data
1981	≤3.34	10.24	≤0.911	5.12	≤0.162	≤0.356
1982	≤0.82	13.00	0.360	9.10	≤0.084	≤0.0227
1983	≤1.12	9.41	≤1.30	0.396	≤0.227	0.0044

Table III: Average Annual Radioactivity Concentrations in Ground Waters (pCi/ℓ)

Year	α	β,γ	$H_3 (x10^3)$
1979	≤7.94	<25	<2
1980	≤0.918	≤ 3.46	<2
1981	≤0.782	≤ 3.72	≤2.03
1982	≤0.955	≤ 0.555	≤2.04
1983	≤0.609	≤ 3.37	≤0.240

Table IV: Average Annual Radioactive Concentrations in Vegetation (pCi/gm)

Year	α	β,γ	Sr-90	I-131	Cs-137	Co-60
1979	0.01854	4.96	0.0995	<0.051	<0.059	<0.0521
1980	<0.00375	4.01	0.017	<0.054	<0.0083	<0.0310
1981	<0.0137	2.71	0.029	<0.12	<0.0178	<0.0017
1982	<0.0196	5.11	0.0169	<0.079	<0.0127	<0.0288
1983	<0.0134	20.3	0.0191	<0.076	<0.1010	<0.0566

Table V: Average Annual Cloud Gamma Monitor results (mRem/yr)

Year	South	East	North	West Boundary
1979	3	3	26	6
1980	0.5	0.5	4.3	1.0
1981	0	5	10	4.1
1982	0	3.2	4.2	0
1983	4.2	17	23	14

Table VI: Average Annual Radioactive Concentrations at Environmental Air Stations (μCi/cc)

Year	Station 1 (51)			Station 2 (52)			Station 3 (53)			Station 4 (54)		
	α	β,γ	I-131	α	β,γ	I-131	α	β,γ	I-131	α	β,γ	I-131
	$(\times 10^{-15})$	$(\times 10^{-14})$	$(\times 10^{-13})$	$(\times 10^{-15})$	$(\times 10^{-14})$	$(\times 10^{-13})$	$(\times 10^{-15})$	$(\times 10^{-14})$	$(\times 10^{-13})$	$(\times 10^{-15})$	$(\times 10^{-14})$	$(\times 10^{-13})$
1979	< 3.5	8.2	< 4.9	< 3.5	10	<4.9	< 3.5	7.4	<5.0	<3.3	14	<5.1
1980	< 2.7	< 6.7	< 4.85	< 2.9	< 7.03	<4.82	< 2.7	<7.04	<4.88	<2.6	<7.68	<4.88
1981	< 3.7	13.8	< 4.86	< 3.9	13.6	<4.87	< 3.9	14.7	<4.93	<3.9	4.5	<4.90
1982	< 1.9	< 2.3	< 4.80	< 1.98	< 1.98	<4.77	< 2.11	<3.22	<4.81	<1.97	<2.63	<4.77
1983	< 2.2	2.4	< 4.20	< 2.3	< 3.3	<4.58	< 2.3	<2.3	<4.25	<2.2	<2.9	<4.78

Table VII: Summary of Environmental Data - 1957

Air Samples

#	α(μCi/cc)	β,γ (μCi/cc)
1.	1.9×10^{-14}	1.3×10^{-13}
2.	1.8×10^{-14}	1.5×10^{-13}
3.	1.1×10^{-14}	1.6×10^{-13}
4.	1.1×10^{-14}	1.4×10^{-13}

Receiving Water

Average for all samples: $\alpha = 1.0\times10^{-9}$ μCi/Ci
$\beta,\gamma = 2.3\times10^{-8}$ μCi/Ci

Ground Water

Average for all samples: $\alpha = 6.3\times10^{-10}$ μCi/Ci
$\beta,\gamma = 6.2\times10^{-9}$ μCi/Ci

Stream Bottom

Average for all samples: $\alpha = 3.2\times10^{-6}$ μCi/gm
$\beta,\gamma = 1.1\times10^{-5}$ μCi/gm

Vegetation

Average for all samples: $\alpha = 4.6\times10^{-6}$ μCi/gm
$\beta,\gamma = 2.6\times10^{-4}$ μCi/gm

23. Question

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.

Answer

The reactor can continue to operate safely for the requested license renewal period. The NTR basically is quite simple and is inherently safe. For instance, no emergency cooling system is required; and failures in the primary coolant pump and piping may occur without any significant safety consequences.

Reliable components are used for the reactor and its support facilities, and maintenance and surveillance programs are in place to insure that continued performance is satisfactory. If unsatisfactory performance is detected, there is no component that cannot be repaired or replaced. For example, the control console has been completely rewired at least two times, and the core container, including the core support structure, also has been replaced two times.

Reactor instrumentation, safety and control rods are inspected, tested and maintained regularly to evaluate current, and predict future, performance by observing trends.

Components also are replaced as part of a facility upgrading program. In 1983 the picoammeters monitoring the neutron flux were replaced. In 1984 the stack gas instrumentation was replaced. Also, during 1984 the stack particulate instrumentation and the remote gamma monitoring instrumentation will be replaced. It is planned to replace the Source Range Monitor and the primary flow transmitter during 1984 or 1985. Control room recorders will be replaced during 1985.

The reactor fuel elements have performed satisfactorily for over 26 years. About 8 years ago, the fuel was inspected and observed to be in good condition. Any deterioration would be detected before significant failure occurred. Water purity is maintained, and reactor operating conditions are not severe, e.g., fuel temperature is less than 200°F.

24. Question

Provide an analysis for an unspecified accident that involves crushing (compacting) the reactor core, and, if fission products are released, calculate the dose in unrestricted areas.

Answer

Crushing the reactor is an incredible accident. The only mechanism for this accident would be the ceiling dropping on the reactor. The ceiling is a three-foot thick reinforced concrete monolith. As can be observed from Figure 3-1, 3-2 and 7-1 in the General Electric Nuclear Test Reactor Safety Analysis Report, NEDG-12727, the reactor is situated between two walls and an alcove. The reactor also contains a 16-inch thick reinforced high density concrete slab over the graphite pack. The ceiling is not likely to fall. However, if it were to fall, it would probably fall as one piece and the reactor would be protected.

Nevertheless, it was assumed that in some unidentified way the reactor graphite pack was fractured to obtain a significantly lower (67% lower) thermal conductivity. The loss of coolant transient analysis (NEDO 12727, Section 11) was rerun and it was determined that peak fuel temperatures increase only 4% to 645°F and remain well below the melting point of the fuel meat. There will be no significant release of fission products from the uranium-aluminum alloy meat if the temperature remains below the eutectic (approx. 1180°F).

Compaction of the fuel would not cause the reactor to go critical since water loss, increased self-shielding in the fuel, and the geometry change due to flattening of the cylindrical core are all negative reactivity effects. Therefore, regardless of the mechanical damage to the reactor or the reactor fuel, there would be no significant release of fission products. In addition, the nature of the fuel (metal alloy) precludes the generation of particulates such as may be encountered with oxide fuels.

25. Question

Provide a new Technical Specification for the linear power channel scrams at 125 kW and perhaps a visible/audible alarm for the Log N at a power of 140 kW. Include provisions for exceeding the normal scram trip point quarterly for purposes of calibration; for example, "the normal power level may be exceeded for periods of less than ______ hours during quarterly instrument and channel calibrations." The table should be footnoted to the effect that trip points are based on the most recent channel calibration.

Answer

Analyses in the General Electric NTR Safety Analysis Report, NEDO-12727 determined that the consequences of accidents with a 150 kw scram point are acceptable. Additional analyses determine that with a maximum 76¢ potential excess reactivity, the consequences of accidents with or without scram are acceptable.

Nevertheless, a revised proposed Technical Specification 2.2.3, Table 3-1 and Table 3-2 are included to reduce the linear power

channel scram to 125 kw and to change the log power channel to a backup alarm at 140 kw.

26. Question

Submit upgraded Technical Specifications using ANSI/ANS 15.1 (1982)

Answer

Revised Technical Specifications will be submitted under separate cover.

27. Question

Submit a list of major modifications made to the GE NTR since 1969.

Answer

The major facility modifications since 1969 are listed below:

1. Enlargement of the penetration for the neutron beam through the reactor cell north wall. (1969)

2. Installation of a rod block circuitry. (1969).

3. Addition of a penetration to the south cell east wall. (1971).

4. Modification of the ventilation system to include new facility areas and increase the capacity from 1,000 cfm to 3,000 cfm. (1972)

5. Redesign of The Manual Poison Sheets. (1975)

6. Installation of a new facility for performing neutron radiography of irradiated materials and an enclosure (north room) added north of the reactor cell. (1976)

7. The horizontal cavity through the graphite was bored out from 3-inches to 5-inches and then refitted with a 3-inch ID sleeve 40-inches long and centered in the graphite pack (1976).

8. Installation of permanent fixed air sample stations at the facility. (1977).

9. Installation of positive latches on the Manual Poison Sheets. (1977)

10. Addition of seismic restrains to the control rod support assembly, fuel loading tank, and reactor shield wall. (1977)