UNITED STATES DEPARTMENT OF COMMERCE
National Bureau of Standards
Washington, D.C. 20234

April 12, 1982

Mr. James R. Miller, Chief
Standardization & Special
Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Miller:

Subject: Response to NRC Questions on NBSR-SAR and Proposed Changes to NBSR Technical Specifications, Docket No. 50-184

Enclosed are the answers to the questions you sent in your letter of December 28, 1981 requesting additional information regarding the application by NBS for license renewal and power increase. Also enclosed, as per our recent discussion, are minor corrections and revisions to previously submitted proposed changes to NBSR Technical Specifications.

Sincerely,

Robert S. Carter

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Chief, Reactor Radiation Division

Enclosures

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ANSWERS TO NRC QUESTIONS ON
NBSR-SAR
Ref: Letter J. R. Miller NRC to R. S. Carter NBS
Dated December 28, 1981

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1. Q. Is it possible to dump radwaste inadvertently into the sewer system from the retention system by a single-point failure?

   A. It is not possible to inadvertently dump radwaste into the sewer system from the retention system by a single-point failure. Additional details are found in answers to question No. 2.

2. Q. Describe the current liquid radwaste handling and monitoring procedures.

   A. a. Relatively high concentration liquid radwaste is saved in containers for eventual disposal as solid waste.

   b. All other suspect liquid waste goes into a 1000-gallon retention tank (located in a concrete pit where it can be visually inspected) and is monitored enroute for radioactivity.

   c. The monitoring instrument, a beta-sensitive G-M counter, alarms (in the Control Room and H. P. Office) at about 2 x background.

   d. When the retention tank is full, its contents are automatically pumped to a 5000-gallon holdup tank (in the pit). In the event of an alarm, the retention tank can be isolated and sampled; transfer to the larger tank could then be made only in the manual mode, by opening a valve and activating a pump.

   e. The 5000-gallon holdup tank must be sampled and the analysis completed before its contents are released to the sanitary sewage system. The control system for routing and sampling waste is located in the Cold Lab basement.

   f. Samples are analyzed for gross alpha, gross beta, and tritium. If the levels are greater than 10% of the 10CFR20.303 limit, supplementary sampling and/or analysis will be done.

   g. In order to release the holdup tank contents to the sewer, it is necessary to operate a valve and activate a pump. The entire dump normally takes about two hours. Releases to the sanitary sewer are adjusted if necessary so as to limit the amount of activity and make effective use of the Bureau's large dilution.

3. Q. Describe how fuel element leaks are detected.

A. A leak from a fuel element would be detected promptly by the fission products monitor located in the helium sweep system. This is followed by sampling and analysis of the helium sweep gas and if necessary by analysis of primary system water. The fission products monitor has proved to be sensitive in reliably detecting even small leaks.

4. Q. What indication does the reactor operator have that the process room is locked?

A. A green indicator light on the Console indicates that the process room door is closed and an annunciator, if clear, indicates that the door is locked. When the door is unlocked, the annunciator sounds and flashes. When the door is opened, the green indicator light goes off and a red indicator light comes on.

5. Q. Describe the changes in process instrumentation that will accompany the proposed upgrade to 20-MW operation.

A. The following changes in the process instrumentation will be made prior to conversion to 20 MW.

a. Flow instrumentation:
   1. Rerange transmitters and scales
      Reactor Outlet - two instruments
      Reactor Inner Plenum
      Secondary Outlet
   2. Replace transmitter and scale
      Reactor Outer Plenum

b. Temperature Instrumentation
   1. Rerange transmitters and scales
      Reactor Inlet Temperature Recorder
      Reactor Outlet Temperature Recorder
      Cooling Tower Temperature Indicator
      Secondary HE-1A Outlet Temperature Indicator
      Secondary HE-1A Inlet Temperature Indicator

   2. Replace Meters
      Primary HE-1A Outlet Temperature Indicator
      Cooling Tower Temperature Indicator
      Secondary HE-1A Outlet Temperature Indicator
      Secondary HE-1A & B Inlet Temperature Indicator

   3. Install New
      Primary HE-1B Outlet Temperature Indicator
      Secondary HE-1B Outlet Temperature Indicator

c. Thermal Power Recorder will be rescaled from 0-15 MW to 0-30 MW.

6. Q. How well did the temperature predictions for the thermal shield compare with the measured values? How will these temperature values change with the proposed increase in power level? (NBSR 9 Section 10.1.2.2)

A. About 20 thermocouples were placed in the reactor shield during construction to monitor the temperature of the thermal shield. They were located near the reactor side of the lead face, at the lead-steel interface, and at the steel-concrete interface. The highest temperature recorded in the lead was 139^{o}F which is somewhat less than the maximum calculated temperature of 150^{o}F given in NBSR 9. Similarly, the highest measured temperature in the steel was 123^{o}F compared to the calculated value of 145^{o}F. Measurements were made at reactor powers of 1, 5, and 10 Mw. These data can be used to extrapolate to 20 Mw operation, yielding temperatures of 205^{o}F and 175^{o}F for the lead and steel respectively.

These numbers are conservative. The thermocouples are located in a very high radiation field so they are heated not only by the surrounding material, but also by the radiation absorbed by the thermocouple itself. Unless the thermal contact between the material and the thermocouple is very good, the thermocouple will be at a higher temperature than the surrounding material resulting in too high a temperature reading.

The thermal shield cooling system is conservatively designed. The flow rate is about 650 gpm and, at 10 MW, the temperature rise of the coolant (H_2O) is about 2^{o}F. At 20 MW, the flow will remain the same, and the temperature rise of the coolant will be about 4^{o}F.

7. Q. How frequently are the shim safety rods changed? What is the expected change frequency at 20-MW operations? (NBSR 9 Section 4.2.1.7)

A. NBSR shim safety arms have been changed only once since construction of the facility. At the time of the change in late 1979 the total megawatt-days were about 25,000. On this basis, it is expected that at 20 MW NBSR shim safety arms would have to be changed about once every 5 years.

8. Q. What is the estimated cadmium lifetime of the poisoned hold-down tubes? Are there any current plans concerning replacement? (NBSR 9, Section 4.4.2)

A. Calculations have been made to estimate the rate of burnup of the cadmium poison tubes. The two-dimensional, two-group, neutron diffusion code EQUIPOISE-3* was used to calculate the neutron absorption rate in the lower 10 cm of cadmium where the neutron flux is greatest. The neutron absorption rate in the 9 cm diameter, 1 mm thick cadmium tube was estimated to be 5.4×10^{13} n/s for 20 MW operation. At this rate 95 years would be required to burn out all the cadmium-113 in the lower 10 cm of the tubes. Therefore, there are no current plans to replace the poisoned hold-down tubes.

*T. B. Fowler and M. L. Tobias, ORNL-3199 (1962).

9. Q. What concentration of tritium is normally found in the spent fuel storage pool? What is the maximum tritium level that has been measured?

A. The storage pool tritium concentration was within 10% of 0.11 μCi/mL in 1981. The maximum concentration for the period 1974 to present was 0.12 μCi/mL in November, 1981. Previous concentrations have been as low as 0.06 μCi/mL (1976 to early 1979).

10. Q. Describe the current fuel element design and core loading configuration, discussing the differences and the advantages from that originally delineated in the 1960-1962 documentation. (NBSR 9, Addendum 1, p. 2-15).

A. The only change in the fuel element design from the original detailed on p. 4-38 and p. 4-39 of NBSR 9 is the removal of some aluminum from the unfueled center sections of the fuel plates. The fuel element is basically the MTR type with 17 fueled plates and two unfueled outside plates. The original fuel plates differed from the conventional MTR plates in that, although the plates were continuous, each plate had two fueled sections separated by a 7" central unfueled section as shown on p. 4-39 of NBSR 9. The current modification replaces each of the original continuous plates by two shorter plates. The fueled sections are 1/2" from the end of the plates and the plates are separated by a 6" gap so the fueled sections are still separated by 7", but there is no aluminum in the central 6" of the gap. The advantage of the current design is that a significant amount of aluminum has been removed from the central gap of the core resulting in a decrease in the aluminum gamma rays entering the beam tubes. Furthermore, the decreased neutron absorption resulting from the removal of the aluminum in the gap contributes to improved core lifetime. All the unfueled side plates, of course, remain continuous.

The current core configuration and U-235 loading are also different from that described in NBSR 9. NBSR 9 described a 24 element core. During startup (December 1967), it was found that the core did not have sufficient reactivity with the 170 g elements available at that time. Therefore, six additional elements were added in the six extra grid positions that had been provided for such a contingency. The original 24 positions shown in figure 4.2 of NBSR 9 remained the same and the six additional elements were placed in the dotted positions in the corners of the hexagon as shown in figure 4.2 of NBSR 9 to achieve the current configuration shown in figure 2.9 of NBSR 9, Addendum 1. This configuration has remained unchanged since reactor startup.

The fuel loading has been increased several times since initial operation. At the present time the fresh element loading is 300 g of U-235. The use of the more heavily loaded elements has lead to improved burnup thereby greatly increasing the lifetime of each element.

11. Q. Describe the improvements and modifications to the primary cooling system that "make maintenance easier and thereby decrease personnel exposures." (NBSR 9 Addendum 1, p. 2-18)

A. The following is a listing of significant improvements and modifications made to the primary cooling system and in the process rooms to make maintenance easier and decrease personnel exposures:

a. Aluminum heat exchangers in primary system have been replaced with more reliable stainless steel ones.

b. All primary drain systems have been redesigned and replaced.

c. Platforms have been installed throughout the process room to make components and piping easily accessible.

d. Additional shielding has been installed around the primary IX columns.

e. Vibration dampers have been installed throughout the primary system piping thereby reducing breakdowns.

f. Equipment has been installed over the primary pumps to facilitate more rapid removal and installation of pump motors.

12. Q. What concentrations of tritium are routinely found in the process room during refueling and maintenance outages? How are personnel exposures controlled?

A. Routinely, tritium levels in the process room are near background levels during refueling and maintenance outages. The process room is usually monitored for tritium by two independent and alarmed monitors with indications in the control room and in the process room. All activities that involve possible significant tritium exposure are usually carried out under conditions of a radiation work permit which sets requirements for exposure limits, monitoring, special equipment and more frequent urine samples plus other conditions as appropriate.

13. Q. What are the probability and potential consequences of coolant boiling after operations at 20 MW? (NBSR 9, Section 4.7.9 and NBSR 9A, p. 1-3)

A. The probability of coolant boiling after operation at 20 MW is very low. No boiling would occur if either shutdown pump operated properly. There are two shutdown pumps each equipped with both an AC and a DC motor. The AC motors can receive power from the commercial grid or backup diesel generators. The DC motors run directly off a backup battery bank. Therefore, it is highly unlikely that both shutdown pumps would fail simultaneously. If, however, such a remote event should occur, it is possible that

boiling might be observed. Studies on the ORR,* which is very similar in fuel element design and fuel plate heat flux for a given power, showed that boiling was not observed after shutdown from 15 MW with no shutdown flow, but was observed after shutdown from 17.5 MW. No damage resulted from these tests. An important difference between the ORR and the NBSR is that the ORR fuel elements are close packed whereas the elements in the NBSR are separated by at least four inches of water. Thus, after shutdown, the NBSR elements are cooled externally and internally whereas the ORR elements are only cooled internally. Therefore, it is possible that boiling would not be observed in the NBSR even at 20 MW although it was observed at 17.5 MW in the ORR. Even if boiling should occur resulting in removal of water from the interior of some fuel elements, the fuel plates would still be adequately cooled by the water surrounding the element as shown in NBSR 9, Addendum 1.

14. Q. Provide a copy of a recent biological shield radiation survey performed while operating at a power level of 10 MW.

A. See enclosure 1, a report of a circumference shield survey of the experimental floor, emphasizing beam-tube locations.

15. Q. Summarize the results of the study of spent fuel temperatures during travel through the transfer chute. (NBSR 9, Section 7.2.1.6 last paragraph)

A. The most significant spent fuel temperature measurement was the one made of the hottest element 8 hours after shutdown from 10 MW. All other measurements were made on elements with less decay heat as a precaution in order to proceed to the next series of measurements with higher decay heat.

Eight hours after shutdown from 10 MW, the hottest element in the NBSR (estimated original operating power of about 435 KW) was placed in the transfer lock completely dry and without auxiliary cooling. The maximum temperature measured (which occured more than two hours after the element was placed dry in the lock) was about 550°F. Introduction of helium flow reduced this temperature by several hundred degrees almost instantaneously. During routine fuel transfer an element is usually dry no more than 15 minutes.

*T.E. Cole and J. A. Cox, "Design and Operation of the ORR," Vol. 10, 86, Second Geneva Conference Report (1958).

16. Q. How much time is required to check for a primary to secondary leak following N-16 monitor alarm? (NBSR 9, Addendum 1, p. 2-23) At what N-16 level is the reactor shut down without waiting for tritium analysis?

A. The N-16 monitor alarms at 200 cpm which is about 4 times background. It takes approximately 5 minutes to obtain and prepare a secondary water sample for tritium analysis. The sample is then counted for 10 minutes or less depending on the activity level. If the tritium analysis confirms the possibility of a leak, the reactor is promptly shutdown. If the N-16 monitor level reaches 1000 cpm, the reactor is immediately shutdown without waiting for a tritium analysis.

17. Q. Describe the improvements in the liquid waste monitoring system. (NBSR 9, Addendum 1, p. 4-5, ECN-242).

A. The improvement to the Liquid Waste Monitoring System provided for the replacement of the old TMC type equipment by modern solid state components. These include a "Ludlum Model 238 count rate meter and a high voltage power supply and recorder. A new coaxial cable was also installed from the count rate meter to the detector.

18. Q. How can it be determined which of the heat exchangers upstream of the N-16 detector is leaking into the secondary cooling system? (NBSR 9 addendum 1, Section 2.6.4 and NBSR 9 Section 1.2.6.6)

A. After the secondary side of a heat exchanger is isolated, repeated samples of the heat exchanger secondary water are taken and analyzed for tritium. If the analysis shows continued increase in the tritium level, then that identifies the heat exchanger as the one leaking.

19. Q. Describe the radiation protection staff. Identify the number, level, and responsibilities of personnel and the lines of communication between them.

A. There are currently three H. P. technicians (H.P.T.'s) one a senior technician (5 years experience), and three health physicists (H. P.'s) all senior (10 years experience), one of whom is a supervisor (S.H.P.) Most routine operations are performed by H.P.T.'s who report anomalies to H.P.'s. New or relatively hazardous operations are monitored by H.P.'s or by H.P.T.'s under direction of H. P.'s. Routinely, lines of communication run from H.P.T.'s to H.P.'s to S.H.P. and back, but direct communication between any staff member and the S.H.P. is appropriate in all cases. Reports (other than for routine surveys) and procedures are written by H.P.'s. In addition NBSR can call on on members of the NBS Health Physics Staff for assistance at anytime.

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

A. Minimum qualifications required:

Position Type (Number)	Usual Qualifications
a. Technician (2-3)	30 sem. hrs. of math, science and /or engineering (or equiv.)
b. Senior technician (0-2)	a. plus 3 years H.P. experience
c. Health physicist (2-3)	B.S. in math, science and/or engineering plus one year experience (or equiv.)
d. Senior H. P. (1-2)	c. plus 5 years experience
e. Supervisory H.P. (1)	d. plus ABHP certification (or equiv.)

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

A. Summary of current radiation safety training for personnel not part of the H. P. staff

Description of Training	Trainees	Frequency, Duration
a. Initial orientation (1) access and security (2) radiation effects (3) regulations and limits (4) ALARA (5) reactor bldg. tour (6) health physics instruments.	New personnel (for unescorted access).	Initial, 2-3 hr.
b. Orientation update: subjects as in a., except (5)	All personnel on access list	Biennial, 1-2 hr.
c. Radioactive waste: (1) segregation and management (2) documentation (3) regulations & other requirements (4) personnel protection	All radioactive waste generators	Annual, 1 hr.
d. Reactor operators: (1) radioactivity (2) Shielding (3) 10CFR (4) other rules & regulations (5) R.A.M. Transfer and disposal (6) radiation detectors (7) inplant & environmental monitoring (8) exposure control & ALARA (9) emergency procedures	License candidates (prep for exam) Requalify	As needed, 30 hr. Biennial As needed

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

A. a. NBSR General Health Physics Procedures:

- HPP 2.1 Radiation Exposure Limits
- HPP 2.2 Personnel Monitoring
- HPP 2.3 Controlled Access Areas
- HPP 2.4 Radiation Work Permit
- HPP 2.5 Obtaining and Using Radioactive Material
- HPP 2.6 Protective Clothing
- HPP 2.7 Contamination Control
- HPP 2.8 Radioactive Waste Control
- HPP 2.9 Removal of R.A.M. from Restricted or Contaminated Areas
- HPP 2.10 Hand and Foot Monitors and Personnel Decontamination

b. Required Routine NBSR H.P. Duties (*1)

(1)	secondary coolant sample, tritium	(*2)
(2)	helium sweep gas fission product sample	(*3)
(3)	neutron film badge change	monthly
(4)	tritium bioassay	monthly
(5)	beta-gamma film badge change	quarterly
(6)	updating form NRC-5	quarterly
(7)	gaseous effluent summary	quarterly
(8)	sealed source leak-testing	quarterly
(9)	protective clothing inventory,respirator checkout	quarterly
(10)	H.P. input to semi-annual report	semi-annual
(11)	review of personnel exposure records	annual

(*1) In addition, there are numerous other activities, routine and special, performed according to the needs of the program, including radiation, contamination and posting surveys, sampling and analysis and instrument quality controls. Frequencies range from semi-weekly to quarterly.

(*2) The secondary coolant is sampled daily whenever the N-16 monitor is down, otherwise monthly.

(*3) Helium sweep samples are taken daily if the fission product monitor is down, otherwise not required routinely.

c. Action Points and Responses:

(1) Dose Rate:
- > 2.5 mrem/hr: pocket dosimeter (PD) required
- > 100 mrem/hr: PD and instrument surveys (IS) during operation
- > 5000 mrem/hr: PD and IS and special administrative authority, prior to exposure

(2) Whole Body Dose: 100 mrem (in 1 day) or 300 mrem (in 1 week)
- (a) Radiation Work Permit normally required
- (b) Increased surveillance by H.P. staff
- (c) Provide special indoctrination, as needed
- (d) Review operational procedure prior to next potential exposure

(3) Contamination (other than personnel): Removal from contamination control zones is not permitted, except for decontamination or disposal, for material contaminated above any of these levels:

Radiation Type(s)	Direct Reading	Removable
Alpha	5 dpm/cm^2	5 dpm/cm^2
Tritium	Not Applic.	100 dpm/cm^2
Other beta-gamma	0.25 mrad/hr	10 dpm/cm^2

(4) Skin Contamination: Decontaminate the skin and obtain medical help if necessary, for contamination above any of these levels:

Radiation Type(s)	Hands	Other than Hands
Alpha, direct reading	1 dpm/cm^2	1 dpm/cm^2
Beta, direct reading	0.1 mrad/hr	0.05 mrad/hr
Alpha/beta, removable	above bkgd	above bkgd

(5) Contamination Control Zones: Post the area as a "Contamination area and require protective clothing, if smear survey results indicate contamination greater than 0.2 dpm/cm^2 alpha or 2 dpm/cm^2 beta activity. "

(6) Tritium Bioassay: Remove the individual from tritium work for the remainder of the week if urinalysis results show a body burden greater than 250 μCi (50% of the amount equivalent to 40 MPC-hours exposure).

23. Q. 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)

A. a. Personnel training is directed toward avoiding any unnecessary exposure, and includes ALARA discussion.

b. Survey instruments are available on all 3 floors of confinement so that workers can do self-monitoring, to supplement H.P. monitoring.

c. Pocket dosimeters are issued to all regular workers in the confinement bldg., and weekly doses are recorded by Health Physics.

d. All exits from confinement to the remainder of Bldg. 235 have hand and foot monitors; their use is required for all personnel. Shoe monitors are available at locations of potentially high contamination levels, for additional contamination control within confinement, providing another barrier to contamination spread.

e. Protective clothing, shielding, and other equipment are provided by Health Physics, and required for some operations.

f. Radiation Work Permits are issued for some operations (see Q.32).

g. Area posting is used for areas even when the level is below that which normally requires posting, if there is potential for significant exposure; e.g., an area reading >0.5 mR/hr with expected occupancy >20 hours/wk.

h. The environmental surveillance system has an expanded quality assurance program, and increased supplementary analyses to evaluate compliance with ALARA.

i. Even relatively low (50-100 mrem) quarterly doses are investigated if received by personnel whose known duties do not involve occupational radiation exposure.

24. Q. For the fixed-position radiation, tritium, and effluent monitors, specify the type of detectors and their efficiencies and operable ranges.

A. See Table I.

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

A. See Table I.

Table I

RADIATION MONITORS

System	Title	Operating Range	Detector Type and Efficiency
RD1.1-7&10	Area Monitors	0.01mr/hr-10R/hr	a
RD1.8&9	Area Monitors	0.1mr/hr-100R/hr	b
RD3.1	Sec. Coolant (N-16)	10^1-10^6CPM	c
RD3.2	He Sweep (Fission Prod.)	10^1-10^6CPM	b
RD3.4	Irradiated Air	10^1-10^6CPM	d
RD3.5	Normal Air	10^1-10^6CPM	c
RD4.1	Stack Air	10^1-10^6CPM	e
RD3.7A	Building Tritium	0 to 6000 M.P.C.	f
RD3.7B	Process Room Tritium	0 to 6000 M.P.C.	f

a. DH1 Tracerlab G.M. tube calibration with Cs-137 & Co-60 sources, detector is $\pm$15% from 0.1 to 2.1 MeV.

b. 18509 Tracerlab G.M. tube calibration with Cs-137 & Co-60 sources, detector is $\pm$15% from 0.1 to 3.0 MeV.

c. 912NB3 Amperex/71916 LND G.M. tube sensitivity is 2.4×10^{-8} μCi/cc/CPM.

d. 1114 Tracerlab G.M. tube for irradiated air sensitivity is 1.3×10^{-7} μCi/cc/CPM.

e. 1114 Tracerlab G.M. tube for stack air sensitivity is 3.5×10^{-8} μCi/cc/CPM.

f. V.R.E. Cary ionization-chamber with a minimum sensitivity of 2×10^{-6} μc/ml for tritium.

RD1.1 through 10 and RD3.1 and 2 are source checked monthly and calibrated annually.

RD3.4 and 5 and RD4.1 trip function is checked prior to startup following a shutdown longer than 24 hours or quarterly and calibrated annually.

RD3.7A and B are calibrated on alternate years.

Annual calibration involves testing the entire system with either a built-in source (as for the Area Monitors) or electronic simulation for the remaining systems, plus an independent source check on all systems.

26. Q. For the radiation monitors that are alarmed, specify the alarm set-points and indicate the expected staff response to each alarm.

A. Area monitors in building are set to alarm at 2.5 mR/hr. Two area monitors in process room are set to alarm at 1 and 10 R/hr respectively.
Response: -Check area monitor board to determine which detector alarmed and the radiation level
-Notify Health Physics
-Survey the area to determine cause of alarm

b. Secondary Cooling (N-16) monitor is set to alarm at 200 cpm.
Response: (see answer to question #16)

c. Helium Sweep (Fission Product) monitor is set to alarm at 10,000 cpm above full power background.
Response: Draw a gas sample and analyze for fission products. If analysis indicates increase in fission gasses the reactor is shut down and search for a leaking fuel element is initiated.

d. Tritium Monitor is set to alarm at 50% of the range set on the recorder (initial alarm at 3 MPC)
Response: -From the recorder determine the area with high tritium activity and if the indicated concentration calls for it, evacuate the area
-Inspect alarmed area and isolate or stop any leak

e. Three effluent monitors that initiate a major scram are set at the equivalent of 10 MPC (Normal and irradiated air at 125,000 cpm stack at 50,000 cpm). A major scram results in shutdown of the reactor and sealing and isolation of the building.
Response: -Confirm that the reactor has scrammed and that the building is sealed and isolated. Determine source and area of activity and take corrective action to reduce or eliminate the activity if possible.
Proceed with appropriate emergency procedure including evacuation if necessary.

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

A. The minimum number of currently available H.P. instruments:
4 very high (to >100 R/hr) gamma survey meters
3 high (>1R/hr) and/or moderate (> 100 mR/hr) gamma survey meter
5 low (to 20 mR/hr) beta-gamma survey meters
1 thermal/fast neutron (to 5000 n/cm^2-sec) counter
2 stationary proportional (alpha & beta) counters
3 hand and foot monitors (beta)

b. Calibration is quarterly for gamma survey meters, monthly for proportional counters, semi-annual for others. The primary calibration sources are Cs-137, Co-60, and Pu-239-Be. Beta response is checked but beta calibration is not routinely done for survey meters.

28. Q. If you anticipate that additional or specialized Health Physics instrumentation may be readily available from other NBS facilities, indicate the type, number, and range of the available equipment.

A. If necessary, the other NBS H.P. program could provide at least two each of high (to > 100 R/hr) and low (to 20 mR/hr) beta-gamma survey meters and one each neutron and alpha survey meters and proportional counter.

29. Q. Describe your personnel monitoring program, including bioassay and in vivo counting capabilities.

A. a. Film badge dosimetry, beta-gamma and neutron is provided for all personnel whose duties normally involve work in the Reactor Building.

b. Tritium urinalysis, routinely, monthly is used for personnel with potentially significant tritium exposure.

c. Whole-body counts, performed at the National Institutes of Health are made for persons with potential for intake of fission or activation products.

d. Pocket ion chamber supplementary dosimetry is used for personnel working in radiation areas.

e. Extremity TLD's for special operations involving potentially high hand or finger doses.

f. Eye examinations are given to personnel with potential for exposure to neutron beams.

30. Q. Describe any routine Quality Assurance studies of the film badge supplier's service.

A. A few films from each lot are exposed to known doses, using NBS standard sources, and supplier-reported doses are compared to known doses delivered. Furthermore film badges are equipped with TLD's that can be promptly counted if warranted.

31. Q. 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.

A. The primary administrative exposure limits are outlined in Q.22, section c. A discussion of constraints is given in Q.32.

32. Q. 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 (SWP's) are used for these events, discuss the applicable requirements, limitations, and approvals.

A. a. Experiments are approved by Chief, Reactor Radiation Division, after review and approval recommendation by the Hazards Evaluation Committee. One-of-a-kind operations, either experimental or maintenance, are coordinated with Health Physics, if there is potential for high exposure. Health Physics then normally issues a Radiation Work Permit (RWP). The RWP form is given as enclosure 2.

b. Examples of operations likely to require an RWP:
(1) Extended occupancy of a high radiation area (as per 10CFR20).
(2) Possible whole body dose in excess of 100 mrem in 1 day or 300 mrem in 1 week.
(3) Extended work in a contamination control zone with levels in excess of 20 dpm/cm^2 alpna or 200 dpm/cm^2 beta.
(4) Exposure to airborne concentrations greater than 10% of MPC.

33. Q. Provide a summary of the NBSR annual personnel exposures (the number of persons receiving total annual exposure within the designated exposure ranges, similar to the report described in 10 CFR 20.407(b)).

A. Summarized annual whole-body doses of NBSR radiation workers are as follows (*1):

Calendar	Number of Workers with Yearly Dose in R (rem):						
Year	.00-.10	.10-.25	.25-.50	.50-.75	.75-1.0	1.0-2.0	>2.0
1970	9	8	4	1	0	0	0
1971	8	2	3	6	5	0	0
1972	2	4	5	3	2	8	2(*2)
1973	9	4	13	0	0	0	1(*3)
1974	10	5	1	4	0	8	0
1975	6	12	7	5	0	0	0
1976	3	11	14	1	1	1	0
1977	4	13	11	2	1	0	0
1978	6	14	9	2	0	0	0
1979	8	10	5	8	0	0	0
1980	6	14	4	2	3	2	0

(*1) This does not include clerical, administrative and support personnel not normally working with radiation, whose doses are typically less than 0.1 rem per year.

(*2) Highest whole body dose: 2.2 rem

(*3) Whole body dose: $\leq$6 rem

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

A. Primary water purification system filters are changed under conditions of a Radiation Work Permit issued by Health Physics and approved by the Reactor Supervisor. The RWP sets the radiation protection requirements and monitoring applicable at the time of change. The filters are usually changed when there is a significant reduction in the flow to the IX columns. Prior to the change the filters are isolated and blown dry and then replaced.

None of the absolute and charcoal filters in the reactor ventilation system have been changed since they were originally installed. Since then, filter activity has been low and filter efficiency continues to be excellent. Pre-filters used to protect the absolute filters from dust and other items in the air are replaced periodically. Installed gauges measure the change in pressure across the filter and the filter is usually changed when the pressure differential is about 3 times the original. Filter replacement is carried out following surveillance by Health Physics and if necessary under their supervision.

35. Q. Describe the gaseous and air particulate effluent sampling equipment. Include location, stack flow rates, sampling rates, and probe descriptions.

A. a. Radioactive gas, primarily argon-41, is measured by a detector located in the stack. Particulates are sampled by a low flow rate isokinetic sampler (flow rate adjusted electronically) and with glass and charcoal filters (counted monthly). The gaseous monitor is calibrated monthly by use of grab samples analyzed for argon-41. Tritium is evaluated by monthly grab samples.

b. The stack monitor is a thin-walled G-M detector located in the stack about two-thirds of the way up. The stack flow rate is approximately 28,000 CFM, the sampler 1.3 CFM.

36. Q. In the equation for mass evaporation rate (M), define the term 0.75 s. (NBSR 9, Addendum 1, p. 3-16).

A. The term (1 + 0.75 s) accounts for the dependency of the mass evaporation rate on the velocity of the air moving across the water surface. The symbol "s" represents this velocity in units of meters per seconds. In the explanation of symbols given at the bottom of p. 3-16 in NBSR 9, Addendum 1, the symbol "V" should be replaced by "s."

Enclosure #1

RADIATION SURVEY - DOSE RATE LEVELS	DATE 12-17-81	NAME 7m+dsh

LOCATION	SPECTROMETER SET	EXPOSURE RATE
BT-1	COARSE	General area 0.8 mr/hr γ " " 0.15 mrem/hr n @ Rope - 0.5 mR/hr γ " " 0.1 mRem/hr n
BT-2	CLOSED	@ SPEC. 0.5 mR/hr γ " " 0.2 mRem/hr n @ Rope - 0.3 mR/hr γ " " - 0.15 mRem/hr n
BT-3	OPEN	@ HOLE - 7 MR/hr γ " " - 6 mRem/hr n @ Rope - 0.5 mR/hr γ " " - 0.6 mRem/hr n
BT-4	OPEN	@ SPEC. - 1.7 mR/hr γ " " - 0.8 mRem/hr n @ Rope - 0.4 mR/hr γ " " 0.55 mRem/hr n
BT-5	CLOSED	@ SHIELDING - 0.7 mR/hr γ " " - 0.45 mRem/hr n @ Rope - 0.5 mR/hr γ " " 0.4 mRem/hr n
BT-6	OPEN	@ SPEC. - 5.2 mR/hr γ " " 2.0 mRem/hr n @ Rope 0.9 mR/hr γ " " 0.5 mRem/hr n
BT-7	CLOSED	General area 0.2 mR/hr γ @ Rope - 0.65 mR/hr γ " " - 0.25 mRem/hr n
GT-2W	OPEN	@ SHIELDING 4.0 mR/hr γ @ Rope 0.3 mR/hr γ " " 0.1 mRem/hr n
BT-8	COARSE	@ SPEC 7.0 mR/hr γ @ Rope - 1.4 mR/hr γ " " 0.4 mRem/hr n
BT-9	FINE	General area - 1.4 mR/hr γ " " - 1.8 mRem/hr n @ Rope 0.6 mR/hr γ " " 0.4 mRem/hr n
CT-E	OPEN	@ SPEC. 6 mR/hr γ @ Rope 0.8 mR/hr γ
Thermal Column	OPEN	@ Rope 1.35 mR/hr γ " " 0.85 mRem/hr n

FORM NBS-362 (3-66)

Enclosure #2

U.S. DEPARTMENT OF COMMERCE
NATIONAL BUREAU OF STANDARDS

RADIATION WORK PERMIT

Date and Time — Extended By ______ RWP No. ______

From ______ a.m. To ______ a.m. To ______ a.m.
______ p.m. ______ p.m. ______ p.m.

Location and Job Description ______

RADIATION SURVEY DATA (By Health Physicist)

Code ☆	Location – Distance From Source	Type Radiation	Rate mrem/hr.	Time For ___ mrem	CONTAMINATION SURVEY Type	Meas.	By	Hour	Date
A									
B									
C									
D									

☆ Work Location Marked with Code

PROTECTION REQUIREMENTS

Rad. Health Surveillance: ☐ Start ☐ Intermittent ☐ Continuous ☐ End of Job

SPECIAL INSTRUCTIONS:

- ☐ Standby companion required
- ☐ Monitor breathing zone
- ☐ Tape gloves - booties to coveralls
- ☐ Provide timekeeper
- ☐ Monitor tools and equipment - end of job
- ☐ Monitor area - end of job
- ☐ Provide assistance for removal of clothing
- ☐ Obtain nasal smears
- ☐ Contact Rad. Health before work in new area
- ☐ Lay down paper or polyethylene on work surface
- ☐ Provide suitable containers for heavy water leaks
- ☐ Provide swipes to wipe surfaces
- ☐ Set up contamination enclosure (tent)
- ☐ Set up local exhaust system
- ☐ Job planning meeting required
- ☐ "Dry Run" of job required
- ☐ Leave bioassay sample
- ☐ (See other side for additional information)
- ☐ Set up RCZ, stanchions, rope, signs, laundry bins, radioactive waste containers

PROTECTIVE CLOTHING

BODY	Coverall - 1 pr.		FEET	Cotton Shoecovers	
	Coverall - 2 pr.			Plastic Booties	
	Plastic Suit			Plastic Shoecovers - 1 pr.	
	Paper Suit			Plastic Shoecovers - 2 pr.	
HEAD	Canvas Hood		RESPIR.	Dust Respirator	
	Cotton cap			Assault Mask	
	Plastic Hood			Self Contained	
				Supplied Air	
HANDS	Cotton Gloves				
	Nylon Gloves		PERSONNEL MONITORS	Film Badge	
	Poly. Gloves 1 pr. – 2 pr.			Supplied Film	
	Rubber Gloves 1 pr. – 2 pr.			Pocket Chamber	
	Gauntlets			Dosimeter	
	Surgeons Gloves			High Range Dosimeter	
				Audible Alarm	
EYES	Safety Glasses				
	Eye Shield				
	Protect Cuts				

APPROVALS

	Regular	Special
Rad. Health		
Supervisor		
Chief Nuclear Engineer		
Chief, Reactor Rad. Division		

Name of Worker ☆	Badge	Code	Rate Used	Work Time	TIME RECORD In	Out	In	Out	In	Out	Total	DOSIMETER (mR) In	Out	Total

1 – White - Work Location
2 – Yellow - Rad. Health
3 – Pink - Control Room

☆ Signing indicates that the permit has been read and understood and that it will be complied with.

USCOMM-DC 32377-P66

Proposed Changes to NBSR Technical Specifications

License No. TR-5

Docket No. 50-184

April 1982

Proposed Changes to NBSR Technical Specifications
License No. TR-5
Docket No. 50-184
April 1982

1. General

   Change AEC to NRC wherever it appears.

2. Section 2.2

   Revise entire section except basis as per attached.

3. Section 3.8

   Revise basis as per attached.

4. Section 3.9

   Revise specifications and basis as per attached.

5. Section 5.1

   Combine specifications 5.1a and 5.1b as into a new 5.1a as follows.

   a. A channel test of the confinement closure system shall be performed quarterly. The trip feature shall be initiated by each of the radiation monitors that provides a signal for confinement closure as well as by the manual major scram switch. A radiation source shall be used to test the trip feature of each of the radiation monitors at least annually.

   Renumber Specification 5.1c as 5.1b and Specification 5.1d as 5.1c.

6. Section 5.4

   References: Switch reference 2 and 3.

7. Section 5.7

   Change Specification 5.7d as follows:

   d. Charcoal adsorber banks in the emergency exhaust system shall be in-place tested with Freon or other halogen at least annually to detect leakage paths caused by settling of the media or deterioration of the filter seals. Leaks greater than 1% of the total flow

will be unacceptable and will require that the effected units be repaired or replaced.

Revise basis as per attached.

8. Section 5.10

Revise specification and basis as per attached.

2.2 Limiting Safety System Settings

Applicabiliity: Applies to limiting settings for instruments monitoring safety limit parameters.

Objective: To assure protective action if any of the principal process variables should approach a safety limit.

Specification: The limiting safety system trip settings shall be as follows:

Reactor Power, % (max)	130	
Reactor Inlet Temperature, ^{o}F (max)	127 (rundown)	
Coolant flow, GPM (Min)	Inner Plenum	Outer Plenum
For Operation up to 10 KW	See Note 1	See Note 1
For Operation up to 10 MW	650	2400
For Operation up to 20 MW	1250	4700

Note 1

May be bypassed during periods of reactor operation (up to 10 KW) when a reduction in safety limit values is permitted (Section 2.1 of these specifications).

Section 3.8 Fuel Handling Within Reactor Vessel

Basis

Each NBSR fuel element employs a latching bar which must be rotated to lock the fuel element in the upper grid.[1] Following fuel handling, it is necessary to insure that this bar is properly positioned so that each element which has been moved cannot "wash out" when flow is initiated. Either of two inspection methods may be employed. A periscope can be used for visual inspection. Alternately a pick-up tool can be positioned at or near the top of each element, one at a time. Tests have shown that flow from a primary main pump will raise an unlatched element above its normal position and thus will be detected by the pick-up tool.

Reference:

[1] Final Safety Analysis Report, NBSR 9, Section 7.2.1.2, page 7-3, April 1966.

3.9 Normal and Post Incident Exhaust Systems

Applicability: Applies to the normal ventilation and emergency exhaust system.

Objective: To insure that normal and emergency ventilation equipment and monitors are operational.

Specifications: The reactor shall not be operated unless:

a. The emergency exhaust system is operable including both fans each with at least one operable motor, and both the absolute and charcoal filters.

b. The reactor building exhaust system is capable of filtering exhaust air and discharging this air above the building roof level.

Basis

The potential radiation exposure to persons at the site boundary and beyond has been calculated following an accidental release of fission product activity.(1) These calculations are based on the proper operation of the emergency exhaust system to maintain the confinement building at a negative pressure and to direct all effluents through filters and up the reactor building stack. The emergency exhaust system has been made redundant to assure its operation. Because of its importance, this redundancy should be available at all times so that any single failure would not preclude system operation when required.(2)

The normal reactor building exhaust is designed to pass reactor building effluents through high efficiency particulate filters at least capable of removing particles of 0.3 microns or greater with an efficiency of at least 99% and discharge them above the reactor building roof level. This system assures filtering and dilution of gaseous effluents

before these effluents reach personnel either on-site or off-site.[3] It can properly perform this funtions using various combinations of its installed fans and building stack. Gaseous effluent monitors are required by Section 3.4 of these specifications.

References:

[1] Final Safety Analysis Report, NBSR 9, Addendum 1, Section 3.4, November 1980.

[2] Final Safety Analysis Report on the NBSR, Supplement B, NBSR 9B, Response No. 5, page 5-1, December 16, 1966.

[3] Fianl Safety Analysis Report on the NBSR, NBSR 9, Section 3.6.4, page 3-15, April 1966.

Section 5.7 Post-Incident and Gaseous Waste Systems

Basis

The post-incident gaseous waste system depends on the proper operation of the emergency exhaust system fans, valves, and filters, which are not routinely in service. Since they are not continuously used, their failure rate due to wear or loading should be low. On the other hand, since they are not being used, their condition in standby must be checked sufficiently often to assure that they will function properly when needed. An operability test of the active components of the emergency exhaust system is performed quarterly to assure that each component will be operable if an emergency condition requires use of the system. The quarterly frequency is considered adequate since this system receives very little wear and since the automatic controls are backed up by manual control provisions.[1,2]

The test frequency of the absolute filters has been established as at least annually.[3] This is the same frequency as used at the Savannah River Laboratory for filters subject to continuous air flow. Since the NBS absolute filters in the emergency exhaust system will be idle except during testing, deterioration should be much less critical than for filters subjected to continuous air flow where dust overloading and air breakthrough are possible after long periods of use. Therefore, an annual testing frequency should be adequate in detecting filter deterioration. Absolute filter efficiency is checked by testing in-place, using a polydispersed aerosol of DOP or other suitable substitute.

The test requirement for the charcoal filters in the emergency exhaust system is basically a physical integrity test. It is prudent to verify that the NBSR filters are not installed or operated in such a way as to be damaged or bypassed.[4] Therefore, a Freon, or other halogen gas, in-place leakage test is required annually to detect leakage paths resulting from charcoal settling and deterioration of the filter seals. Experience at the Savannah River plant has demonstrated Freon or other halgen gas to be an acceptable means for determining the leakage characteristics of charcoal filter installations. The 1% acceptability limit is specified to give assurance that a high overall iodine filter efficiency significantly above the 95% used in the DBA will be maintained.[5]

References:

[1] Final Safety Analysis Report, Supplement A, NBSR 9A, Response No. 12, pages 12-10 through 12-17, October 1, 1966.

[2] Final Safety Analysis Report, Supplement B, NBSR 9B, Response No. 5, pages 5-1 and 5-2, December 16, 1966.

[3] Final Safety Analysis Report, Supplement B, NBSR 9B, Response No. 1, page 1-1, December 16, 1966.

[4] Final Safety Analysis Report, Supplement B, NBSR 9B, Response No.1, page 1-1, December 16, 1966.

[5] Final Safety Analysis Report, NBSR 9, Addendum 1, Section 3.4.2, November 1980.

Section 5.10 Environmental Monitoring

Specification: An area monitoring program shall be carried out and shall include as a minimum the quarterly analysis of samples from area streams, vegetation or soil, and air monitoring.

Basis

Consistent with the recommendations of the U. S. Geological Survey, a periodic sampling program of area wells and streams has been conducted since November, 1962.[1] In order to assure more complete sampling of the area surrounding the NBSR, this program has been expanded to include area vegetation or soil samples (soil samples being more meaningful during the nongrowing season). By 1982, most of the wells in the vicinity of the facility had been closed and only one well remained active, making further analysis of well water no longer meaningful. Sampling of area streams however is continuing and is required. Thermoluminescent dosimeters or other devices are also placed around the perimeter of the NBSR site to monitor the air. The continuation of this environmental monitoring program will assure that the operation of the NBSR presents no significant hazard to the public health and safety. In the 15 years since the NBSR began operation, the environmental monitoring program revealed nothing of significance thereby confirming that operation of the NBSR has little or no effect on the environment. The quarterly frequency is considered adequate to detect any long-term changes in the activity levels in the vicinity of the NBSR. Shorter term changes would require a significant release which would be detected by the exhaust system radiation monitors.

Reference:

[1] Final Safety Analysis Report on the NBSR, NBSR9, Section 2.4.3.2, page 2-6, April 1966.