DEFENSE NUCLEAR AGENCY
ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
BETHESDA, MARYLAND 20014

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OCT 9 1981

Mr. James R. Miller
Chief, Standardization & Special
Projects Branch
Division of Licensing
United States Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miller:

Reference your letters dated 2 September 1981, (with list of questions) and 18 September 1981, approval of our request for an extension to answer those questions.

We have completed our answers to the aforementioned list of questions and are submitting twenty copies, with supporting data, in accordance with guidance from your office.

Should you have any questions please contact Major Ronald R. Smoker, Chief, Radiation Sources Division, 295-1096/1290.

Sincerely,

BOBBY R. ADCOCK
Colonel, MSC, USA
Acting Director

Enclosures
as stated

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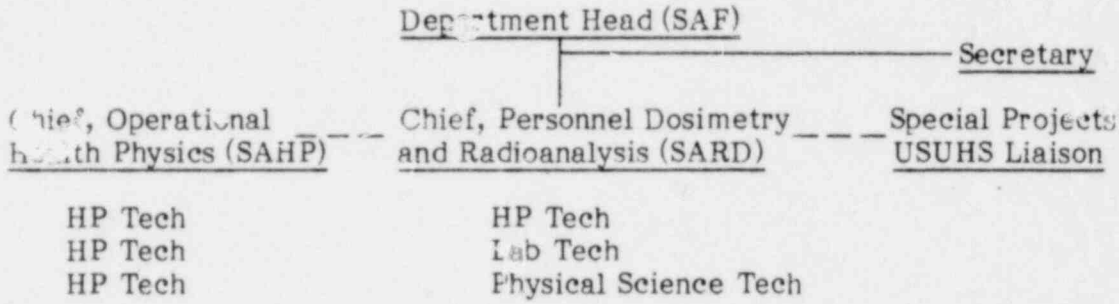
AFRRI ANSWERS TO NRC STAFF QUESTIONS CONCERNING
THE AFRRI SAFETY ANALYSIS REPORT (SAR) DATED MAY 12, 1981

1. DESCRIBE THE RADIATION PROTECTION STAFF. IDENTIFY THE NUMBER, LEVEL, RESPONSIBILITIES OF PERSONNEL, AND THE LINES OF COMMUNICATION.

The following flow diagram describes the Radiation Protection Staff, number, level, and lines of communication.



The responsibilities of the above positions are as follows:

Head, Radiation Safety:

Develops and maintains a comprehensive Health Physics Program, which encompasses all sources of radiation within AFRRI.

Coordinates and verifies compliance with all regulatory agencies (including NRC, DOT, EPA, FDA, and DOD) with regard to restrictions placed on the use of these radiation sources and implementation of the As Low As Reasonably Achievable (ALARA) philosophy. Provides radiological analysis support for all uses of radioactive materials within the AFRRI. Provides Health Physics guidance to command personnel. Directs investigation into unsolved Health Physics problems.

Department Secretary: Assists Department Head administratively in fulfilling his duties.

Chief, Radiation Health Physics (SAHP):

Maintains the direct support of the Health Physics Programs in accordance with the ALARA philosophy, regulatory compliance monitoring program, and certain support activities for these programs. These include performing field calibration procedures, maintaining the environmental monitoring and impact analysis program, providing primary documentation for all health physics monitoring activities, formulating and presenting radiological training programs, and providing all direct monitoring support. Provides health physics guidance to command personnel as requested.

Radiation Health Physics Division Technicians: Assists Chief, Health Physics Division in fulfilling his responsibilities.

Chief, Radiation Radioanalysis and Dosimetry Division (SARD):

Maintains radioanalysis and radiological instrument radiocalibration programs and facilities to meet all the requirements of the Health Physics Programs, the needs of the radiation facilities, and the needs of all research investigators in the Institute. Provides technical assistance to research investigators in radioanalysis methods and procedures. Maintains related records and operates a quality control program as required by the applicable regulatory agencies. Maintains the personnel radiation exposure-monitoring program. Provides special hazard analysis as required and radioanalysis for other government and non-government agencies as requested.

Radiation Radioanalysis and Dosimetry Division Technicians: Assists Chief, SARD in fulfilling his responsibilities.

USUHS Liaison and Special Projects Officer:

Maintains liaision and review of USUHS Radiation Safety program. Investigates unsolved health physics problems as directed. Provides assistance to Chief, Health Physics Division as required. Acts for Chief, Health Physics Division in his absence.

2. OUTLINE THE MINIMUM QUALIFICATIONS (TRAINING AND/OR PREVIOUS EXPERIENCE) FOR EACH OF YOUR HEALTH-PHYSICS-RELATED POSITIONS.

The minimum qualifications (training and/or previous experience) for the Health Physics positions are as follows:

Head, SAF:

BS natural sciences or engineering

3 years professional experience

Experience must have demonstrated a professional knowledge of Health Physics and indicated ability to apply scientific methodology to radiation protection problems. Previous supervisory experience.

Chief, SAHP, SARD, USUHS liaison/special projects:

BS degree, natural science or engineering; or equivalent experience

3 years professional experience

Experience must have demonstrated a professional knowledge of Health Physics and indicated ability to apply scientific methodology to radiation protection problems.

HP Technicians, Lab Technicians:

Associate degree, natural science or engineering or equivalent.

3. DESCRIBE ANY HEALTH PHYSICS TRAINING FOR NON-HEALTH PHYSICS STAFF. IF POSSIBLE, PROVIDE A TOPIC OUTLINE OF THE COURSE, INDICATING THE LEVEL AND DURATION OF EACH COURSE.

The following is a breakdown of health physics training for the non-health physics staff assigned to AFRRI. There are four categories of training provided at AFRRI; three routine and one special (usually requested or initiated by the Safety Department).

a. Initial Radiation Safety Briefing. Given to all incoming personnel who have unescorted access to AFRRI (Duration: ~1/2 hour). The briefing covers basic NRC guidelines, basic radiation protection practices, emergency response, and other items appropriate to the ALARA philosophy. In addition, any questions concerning radiation the individual may have are addressed.

b. Basic Radiation Safety Briefing. This briefing is given to all personnel annually (Duration: 1/2 to 1 hour) and includes a review of the basic health physics practices and AFRRI's radiation protection policies. This briefing normally includes all the material in item a. above.

c. Radioactive Source Users Briefing. This annual briefing (Minimum Duration: 1 hour) is presented to all individuals handling radioactive sources. In addition to the items addressed in a. and b. above, a review of safe handling methods, accountability, disposal, past experiences and other items appropriate to the safe handling of radionuclides are presented.

d. Specialized radiation safety training provided by health physics staff on specialized topics as requested or as necessary, e.g., air mask training, animal handlers handling of radioactive animals, Janitor briefings, etc.

The reactor staff and other selected members in the Institute are trained in radiation detection surveys, decontamination procedures, and radiation protection. The extent of actual training of these individuals varies depending on the experience of the personnel involved.

In addition, outside the reactor staff and health physics staff there are approximately 5-15 military officers assigned to AFRRI at any given time who could be classified as professional health physicists and 5-15 individuals who have had civilian and/or military training in radiation protection who could be called on to provide health physics support if deemed necessary. Also, many investigators at AFRRI have had broad training from previous employment.

Non-AFRRI training. Several organizations have health physics training available for AFRRI personnel. This training is frequently used. For example, AFRRI personnel are routinely directed to attend the National Institute of Health's Radiation Users Course.

4. SUMMARIZE YOUR GENERAL RADIATION SAFETY PROCEDURES. IDENTIFY THE MINIMUM FREQUENCY OF SURVEY, ACTION POINTS, AND APPROPRIATE RESPONSES.

The general radiation safety procedures are a set of radiological safety instructions. They define appropriate responsibilities of both the Safety Department and AFRRI investigators. Also included in the instructions are the charters for the Reactor and Radiation Facility Safety Committee and the Radionuclide and X-ray Sources Facility Safety Committee.

There are many types of surveys which occur at AFRRI depending on the actual on-going work. The most common minimum surveys are as follows:

SURVEY	FREQUENCY	TYPE	RESPONSE
Institute-wide Smears	Weekly	Smear	If levels above 50 pCi/100 cm^2 decontaminate area
Air Particulate Survey	Weekly	Air Sample	Identify source of contaminant and remove: 0.5 pCi/cm^3
Reactor Exposure Room Entrance	As necessary	Whole Body	Safety member determines stay time not to exceed 50 mrem/day/person
Sealed Source Leak Test	Semi-annually	Smear	Remove leaking source if any
Institute Whole Body Gamma	Weekly	Whole Body Gamma	Identify exposed sources and remove.

Specific instructions dealing with the detailed implementation of radiological safety are contained within AFRRI's internal Health Physics Procedures (HPPs).

5. DESCRIBE THE PROGRAM TO ENSURE THAT PERSONNEL RADIATION EXPOSURE AND RELEASES OF RADIOACTIVE MATERIAL ARE MAINTAINED AT A LEVEL THAT IS "AS LOW AS REASONABLY ACHIEVABLE" (ALARA).

The program that exists to ensure that personnel radiation exposure and releases of radioactive material are maintained at a level that is "As Low As Reasonably Achievable" (ALARA) is as follows:

Generally: AFRRI's policy is to exercise and maintain the ALARA program at all times. A Directorate policy is given to all radiation workers at AFRRI in the Radiological Safety Booklet (given at the initial employee safety briefing).

Personnel Exposure: All experiments in the reactor must be done under a routine or special authorization that has been reviewed by the Reactor and Radiation Facility Safety Committee. Each individual specific experiment that is performed under these committee authorizations is reviewed by the Reactor Physicist-in-Charge (or his designee) and a professional member of the Health Physics staff. Evaluation of the personnel radiation hazard is of primary importance. All entries into any radiation field other than that existing from an approved quantity of a specific radionuclide is first approved by a member of the professional Health Physics staff. All entries into the reactor exposure rooms or a high radiation field are preceded by a health physics monitor who controls the stay time so as not to exceed 50 mr/day or 100 mr/week unless prior approval is granted at the level of SAF Department Head or higher authority. All non-routine situations and activities of a planned nature are governed by a special work permit. In addition, the reactor staff oversees all operations for compliance with the ALARA concept.

Radioactive Releases: Gaseous radioeffluent releases are continuously monitored by the stack gas monitoring system. Air in various areas of AFRRI is monitored by various Continuous Air Monitors (CAMs) located at designated areas throughout the complex. In addition, several CAMs are available for use if an unexpected event occurs. Liquid radioactive waste is collected via the warm/hot waste drain system and held in the retention tanks until sampled for radioactive content. All radioactive materials shipped from AFRRI are shipped in accordance with Department of Transportation (DoT) regulations. All areas where radioactive materials are used are posted accordingly. Any radioactive material in excess of limits identified in the radiological safety instructions (RSIs) are retained in warm storage. Spills or areas of contamination are immediately cleaned upon notification except the reactor prep area which, although normally uncontaminated, is treated as a contamination area to maintain control for entry into the reactor exposure rooms.

Operationally: The installation and use of the in-core experimental tube for radioisotope production and activation analysis in place of the pneumatic tube system has resulted in far lower production and subsequent releases of Ar-41 to the environment. Also, design features within the exposure rooms reduce the amounts of Ar-41 produced and released by reducing the thermal neutron population within the rooms, particularly in ER #1. The use of ER #2 is limited operationally since it is the exposure facility in which the greatest amount of Ar-41 is produced per kw-hr. Each experiment is reviewed to insure that exposures and byproduct isotopes produced are as low as possible within the context of the necessary experimental procedures.

6. FOR YOUR FIXED-POSITION RADIATION MONITORS, GAS MONITORS, AND EFFLUENT PARTICULATE MONITOR, SPECIFY THE TYPE OF DETECTORS AND THEIR EFFICIENCIES AND THE OPERABLE RANGES.

a. The Radiation Area Monitors use a plastic gamma scintillator operating in the current integrating mode. Due to the nature of the plastic used and the integration mode, the calibration is, for all practical purposes, a direct indication of the actual mr/hr field in free air. The ranges vary dependent upon location and fields anticipated. Those located in normally low radiation areas are usually 0.05-50 mr/hr and those where high radiation areas

may exist are either 0.001-100 or 0.001-1000 r/hr. Detectors and electronics are such that the monitors will not saturate up to 100X the full scale reading.

b. The stack gas monitor detector is a gas flow proportional counter sandwiched between two air chambers (each of which has a 10 liter volume). The detector efficiency is 7%, and the instrument's range is 0.003 to 3.0 pCi/cc.

c. The stack particulate monitor detector is a pancake GM tube. The nominal detector efficiency for Sr-90/Y-90 is 15%. This instrument has a scale range of 0.1 to 1000 cps with a background reading of 1 cps.

7. FOR THE FIXED-POSITION MONITORS, DESCRIBE THE METHODS AND FREQUENCY OF INSTRUMENT CALIBRATIONS AND THE ROUTINE OPERATIONAL CHECKS.

The Remote Area Monitors (RAM's) are calibrated quarterly using a Cs-137 source of approximately 60 mCi at various distances and dose rates dependent upon the ranges of the monitors. Operational checks of the fixed position monitors are made as part of the regularly scheduled weekly maintenance and as a part of the daily reactor start-up checklist. These checks include verification of alarm setpoints and testing of the high level alarm and failure alarm.

8. FOR MONITORS THAT ARE ALARMED, SPECIFY THE ALARM SET-POINTS AND INDICATE THE STAFF RESPONSES TO EACH ALARM.

This information is contained in RSD Instruction 5-3, dated 27 March 81, which was provided earlier (copy included).

9. IDENTIFY THE TYPE, NUMBER, AND OPERABLE RANGE OF ALL OF THE PORTABLE HEALTH PHYSICS INSTRUMENTS. SPECIFY THE FREQUENCY AND METHODS OF CALIBRATION.

Type	Number	Range
VICT 440	2	Medium
VICT 471	4	Medium
VICT 493	1	Low
VICT 666	1	High
THYAC 11	3	Low
THYAC 111	2	Low
JUNO 7	1	High
HPI 1060	2	High
Teletector	3	High
SADORA	1	High
LUDLUM 2	1	Medium
LUDLUM 3	1	Low
LUDLUM 16	1	Low
EDGERTON SBL55	4	(Aerial Survey)
EBER PAC-18A	2	Low - Alpha
EBER PNR-4	2	Medium - Neutron

Beta/gamma survey instruments are calibrated quarterly with a Cs-137 source. Neutron instruments are calibrated semi-annually using a 10 Ci Pu-Be source. Alpha survey instruments are calibrated semi-annually using a plutonium alpha standard.

10. IF YOU ANTICIPATE THAT ADDITIONAL OR SPECIALIZED INSTRUMENTATION MAY BE READILY AVAILABLE FROM OTHER NNMC FACILITIES, INDICATE THE TYPE, NUMBER, AND RANGE OF THE AVAILABLE EQUIPMENT.

The following additional or specialized instrumentation could be made readily available from other NNMC facilities if needed:

Instrument	Type	Number	Range
AN/PDR-27	GM	10+	Low
VIC 440 RF	Ionization	2	Medium
AN/PDR-43E	GM	5+	High
AN/PDR-56	Alpha	10+	To 1E+6 cpm
AN/PDR-70	Neutron	1	To 1E+3 mrem/hr
Camberra 8180	MCA	1	N/A
Camberra Series 30	MCA	1	N/A
Beckman Wide-Beta	Prop	1	N/A
LS Counter	Scint.	3+	N/A

11. DESCRIBE YOUR PERSONNEL MONITORING PROGRAM, INCLUDING BIO-ASSAY AND IN VIVO COUNTING, IF USED.

See RSI 220, Personnel Monitoring Administration.
RSI 230, Staff Personnel Monitoring.
RSI 240, Nonstaff Personnel Monitoring.
HPP 1-6, Internal Exposure Monitoring Program.
(References attached)

12. IDENTIFY ANY ADMINISTRATIVE EXPOSURE LIMITS AND THE ANTICIPATED ACTIONS IF THESE LEVELS ARE EXCEEDED. ALSO, IDENTIFY THE OPERATIONAL CONSTRAINTS THAT ARE PLACED ON PERSONNEL ENTERING POTENTIAL RADIATION/HIGH RADIATION OR CONTAMINATED AREAS.

AFRRI administrative exposure limits have been set at 50 mrem for one day and/or 100 mrem in one week. Prior approval of the SAF Department Head is required to exceed these limits. Head, SAF, is notified of all exposures in excess of 50 mrem for the quarter in question. If the dose rate in a work area is to exceed 1.5 rem/hr, extremity TLD's are issued. All exposure situations are handled in accordance with the concepts of ALARA (As Low As Reasonably Achievable). Entries into the exposure rooms or removal of items from the CET are monitored by a member of SAHP. Work may continue in a non-high radiation area without the immediate presence of a RADSAF member. If there is extended work in a high radiation area, a member of SAHP will normally be present. If the area to be entered is posted as a contamination area, the proper protective clothing is worn.

13. WHAT IS THE HEALTH PHYSICS REVIEW AND EXPOSURE CONTROL OF ONE-OF-A-KIND, SHORT-TERM, LOW-TO INTERMEDIATE-RISK TASKS, SUCH AS SIMPLE BUT NONROUTINE MAINTENANCE ACTIVITIES AND ONE-SHOT EXPERIMENTAL MEASUREMENTS? IF SPECIAL WORK PERMITS (SWPs) ARE USED FOR THESE EVENTS, PLEASE DISCUSS THE APPLICABLE REQUIREMENTS, LIMITATIONS, AND APPROVALS.

The health physics review and exposure control of one-of-a-kind, short-term, low-to-intermediate risk tasks, such as simple but non-routine maintenance activities and one shot experiments are as follows:

Experiments: All experiments in the reactor must be done under a routine or special authorization that has been reviewed by the Reactor and Radiation Facility Safety Committee. Each individual experiment under these authorizations is reviewed by the Reactor Physicist-in-Charge or his designee as well as a member of the professional Health Physics staff. If no standing routine or special authorization exists, the experimental protocol must go to the committee for review and approval.

SWPs: Special Work Permits are not routinely needed for experiments. Experiments are designed based on the ALARA principle and are reviewed by the reactor staff and the professional Health Physics staff for compliance with the ALARA philosophy. If constraints are such that a planned experiment or activity will involve one of the following, a Special Work Permit may be issued depending upon the circumstances:

a. Extended work in a High-Radiation area.

b. Airborne radioactivity above 10% MPCa.

c. Unsealed alpha source useage.

d. Procedures involving volatile radioactive sources.

e. A credible potential for a radionuclide intake greater than 10% of maximum permissible burden.

The SWP outlines any special actions and/or precautions and is approved by the initiator, and a professional member of the Health Physics staff.

Reactor Maintenance: If there are any maintenance procedures that involve a planned personnel exposure, a member of SAHP will be present for surveying. All work procedures are governed by the ALARA philosophy. Any reactor maintenance or procedure which could involve issuing an SWP is considered non-routine and handled accordingly. Each Monday, at a weekly meeting, the reactor's work schedule for the current week is reviewed by a member of the Health Physics staff and the Reactor staff to determine those areas where health physics support will or may be required.

14. ARE THE CHECK VALVES IN THE PURIFICATION AND COOLING SYSTEM REGULARLY CHECKED FOR LEAKAGE?

Inspection for external leakage from these valves and all other components of the reactor water system located in the upper and lower equipment rooms is an item checked as part of the daily reactor startup and shutdown procedures. The quick-close manual gate valves in the reactor water purification and cooling system are exercised semi-annually as a part of regularly scheduled maintenance.

15. PROVIDE A SUMMARY OF THE AFRRI ANNUAL WHOLE BODY RADIATION EXPOSURES (THE NUMBER OF PERSONS RECEIVING TOTAL ANNUAL EXPOSURE WITHIN THE DESIGNATED EXPOSURE INTERVALS; THAT IS, LESS THAN 0.5 REM, 0.5 TO 1.0 REM, AND SO ON) FOR PAST 10 YEARS OF OPERATION.

The Annual Individual Whole Body Exposures for AFRRI/USUHS

YEAR	Less than 0.5 REM	0.5 to 1.0 REM	1.0 to 3.0 REM
1980	881	0	0
1979	810	0	0
1978	800	0	0
1977	385	0	0
1976	300	0	0
1975	600	2	0
1974	330	5	2
1973	260	0	8
1972	250	5	2
1971	224	3	3

The USUHS is included in data after 1976.

Exposure information includes all sources of radiation within AFRRI/USUHS.

16. WHAT TYPE OF NEUTRON DOSIMETER IS USED FOR PERSONNEL DOSIMETRY? DESCRIBE BRIEFLY THE NEUTRON CALIBRATION OF THE DOSIMETER.

Due to the nature of the work performed at AFRRI, neutron dosimetry is not routinely required for all AFRRI personnel. However, supplemental neutron dosimeters are issued to all personnel who might reasonably be exposed to neutrons. It should be noted that this device is additionally used as a backup dosimeter for all personnel who might receive an annual exposure greater than 200 mrem.

DOSIMETER DESCRIPTION: A Li-6 chip and a Li-7 chip in a Harshaw TLD card (type NG-67) are placed in the AFRRI Dosimeter/ID credential holder to serve as a neutron dosimeter. This dosimetry application is patterned after Hankins (Health Physics, 31,2, page 170, August 1976). This is not an albedo dosimeter which is subject to body distance sensitivity.

DOSIMETER CALIBRATION: The response of the neutron dosimeters is determined for a known neutron field from a Pu-Be source in a low scatter environment. This response factor is used routinely to evaluate the possible neutron exposure. Since this dosimeter is primarily sensitive to the thermal neutron component of the field and was calibrated in a field with a low thermal neutron component, the indicated dose will be greater than or equal to the actual dose. In the event of a significant exposure to neutron radiation, the fields for the exposure would be reproduced and used to recalibrate the dosimeter and thus determine the actual dose for the exposure.

17. WHAT CHECKS ARE MADE DURING THE READOUT OF PERSONNEL DOSIMETERS TO ASSURE THE PROPER OPERATION OF READ-OUT DEVICE?

The instrument used to read out the personnel dosimeters is the HARSHAW MODEL 2271 TLD READER. Each day that TLD's are read all settings on the instrument are checked. The built-in light source and dark current are measured to check the functionality of the instrument. Preselected TLD's are dosed to 100 mrem equivalent gamma exposure using an automatic dose calibrator with an internal Sr-90 source. This information is used to determine the instrument calibration factor. During readout of dosimeters which have been worn by personnel, approximately every 50th card is a standard TLD card which has been exposed to 100 mr from Cs-137 gamma radiation. These results are recorded and if necessary can be used to reevaluate the calculated dosimeter doses to correct for any malfunctions or changes in the instrument response. In addition, the glow curves of all TLD readings are collected and are used to evaluate the operation of the equipment.

18. DESCRIBE THE RADWASTE HANDLING SYSTEMS AND THE APPLICABLE PROCEDURES, INCLUDING METHOD FOR MONITORING AND MEASURING ACTIVITY LEVELS IN RADIOACTIVE WASTE BEFORE FINAL DISPOSAL.

Liquid radwaste is accumulated in the liquid radioactive waste storage tank facility and processed according to the procedures outlined in Health Physics Procedure 6-4, "Waste Tank Facility." Solid waste is compacted, packaged, and handled according to Health Physics Procedure 6-3, "Solid Radioactive Waste." Disposal of solid wastes is under contract to Headquarters, U.S. Army Armament Materiel Readiness Command, Rock Island, Illinois, which in turn has a contract with South West Nuclear Co. for the actual pickup and transportation to a commercial burial ground.

19. WHAT ARE THE TECHNICAL SPECIFICATIONS OF THE CONDUCTIVITY CELLS USED TO MONITOR THE WATER DEMINERALIZER?

All three primary conductivity cells are in-line, flow-through titanium electrodes with a concentric cylinder design, model 920-0.01T, manufactured by Balsbaugh, Inc., Plymouth, Massachusetts. They have a cell constant of 0.01 cm^{-1}. Their readout is a Balsbaugh model 920 digital water monitor controller, temperature compensated to 25°C, with a range of 0-20 MΩ-cm. The entire system is accurate to ± 1%.

A back-up manual system utilizes a Horizon Ecology model 1484-10 battery operated meter with tungsten electrodes, temperature compensated to 20°C by an internal thermistor network. The range of the system is 0-20,000 μmhos/cm in 5 range steps with an accuracy of ± 2% over the full range.

20. WHAT FUNCTIONAL TESTS ARE CONDUCTED ON THE REACTOR ROOM VENTILATION ISOLATION SYSTEM AND THE OVERPRESSURE RELIEF DAMPER? HOW FREQUENTLY ARE THESE TESTS PERFORMED?

The four reactor room isolation dampers are exercised whenever reactor operation is scheduled as part of the normal reactor start-up checklist procedure. Indicator lights on the auxiliary panel in the control room come "on" when the dampers close, and a significant change in air flow becomes apparent. Once each month the closure mechanisms on each of the dampers are visually inspected and tested for correct operation by observing the air solenoid plunger and linkage travel.

The overpressure relief damper is visually inspected during the reactor facility annual shutdown maintenance period.

21. DESCRIBE THE PROCEDURES FOR MONITORING AND CHANGING THE FILTERS IN THE VENTILATION AND WATER PURIFICATION SYSTEMS.

The reactor building ventilation system has an air plenum with three sets of filters located in the stack exhaust line. The plenum contains a set of prefilters, a set of roughing filters, and a set of absolute filters in series. The pre- and roughing filters are monitored by a differential pressure measuring system as one unit. The absolute filters are monitored by another separate differential pressure measuring system. Each of these differential pressure measurements are recorded whenever operation of the reactor is scheduled as part of the normal reactor startup checklist procedure. When the differential pressure reaches a predetermined level, the filters are monitored, removed from the plenum, and stored for disposal. New filters are then placed in the plenum. If the absolute filters are changed, the entire system is checked for efficiency and proper edge seal using standard DOP test procedures.

Contained in the reactor primary water purification system are a set of particulate filters housed in a CUNO canister. The differential pressure across these filters is monitored and recorded (as part of the normal reactor startup checklist procedure) any day that a reactor operation is planned. Upon reaching a predetermined differential pressure, the canister is isolated from the purification system. The canister is then opened, the filters are removed and replaced, the canister resealed, placed back in line and checked for leaks. The old filters are removed for analysis and subsequently processed according to the results of that analysis.

The time interval between filter changes in both the air and water systems varies depending on the operations and operational conditions associated with each system.

22. LIST AND DISCUSS THE FACILITY COMPONENTS THAT ARE ON EMERGENCY BACK-UP POWER. DESCRIBE THE TEST AND MAINTENANCE SCHEDULE FOR THE BACK-UP POWER SUPPLIES.

The only reactor related component on emergency back-up power is the NMC criticality monitor which is provided with a 12-volt battery backup, NMC auxiliary power supply model PSA-11. Monthly testing and maintenance consists of checking the operation of the system while on auxilary power, checking the operation of the battery charging circuit, checking the water level and specific gravity of the battery, and checking the battery for corrosion and loose connections. Industrial type safety lighting units within the reactor area also operate on emergency power; however, the units are not directly related to the reactor system.

23. HOW MANY USED FUEL ELEMENTS ARE IN THE POOL STORAGE RACKS? WHAT PROBLEMS WILL ARISE IF THE WATER LEVEL DROPS? IF THERE IS A TOTAL LOSS OF COOLANT, HOW WILL THE FUEL ELEMENTS BE MOVED TO REPAIR THE LEAK? INCLUDE THE ESTIMATED EXPOSURES OF OPERATIONS PERSONNEL. (IV-14)

At present, there are nine (9) fuel elements in pool storage specifically located within three of the six available fuel element storage racks. Each of these nine spare fuel elements has previously experienced some burnup in the core.

Each fuel element storage rack can accommodate up to twelve (12) fuel elements in a neutronically safe configuration and provides adequate cooling.

Each fuel element storage rack is composed of two subracks which are welded together in a stepped arrangement with the subracks (each of which can hold six fuel elements) at different but fixed elevations. The upper subrack is fitted with two aluminum tie rods which extend vertically above the pool water surface and clamp securely to the reactor tank wall rim.

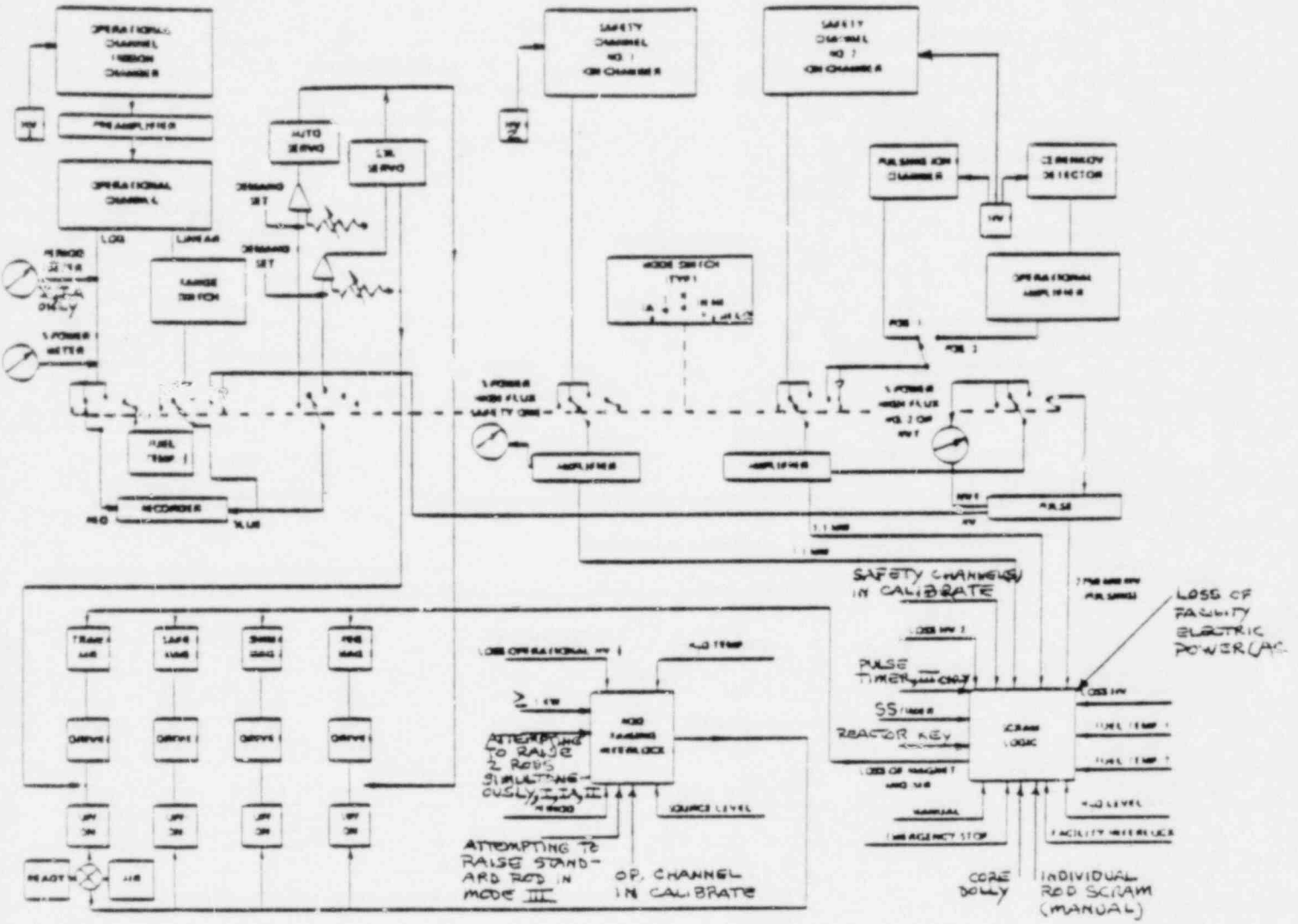
With this arrangement, the upper subrack of each fuel element storage rack lies against the inside surface of the reactor tank under approximately nine feet of water while the lower subrack of each fuel element storage rack is approximately 6 inches away from the inside surface of the reactor tank under approximately twelve feet of water.

In the unlikely event that a partial loss of water from the reactor tank should occur such that the storage racks become partially or totally uncovered but the core remains covered, the fuel elements in storage would be lowered beneath the standing water level either individually, using the fuel element handling tool, or collectively by removing the storage racks having fuel elements in them to the bottom of the pool beneath the standing water level. If the drop in pool water level occurred as a result of a leak or breach within the primary water and purification loop, no further in-pool precautionary measures would be required prior to repairing the leak or breach and recovery of the facility. If, however, the drop in pool water level occurred as a result of a leak or breach in the reactor tank wall, the core and spare fuel elements would be moved into the pool water volume opposite the breach and the lead shield doors would be closed to provide the maximum amount of water and lead shielding between the breach point and the fuel. Recovery operations in either of these two cases would involve only slight occupational exposures most likely on the order of 10-50 mrem but certainly less than 100 mrem to any single individual.

In the extremely unlikely event that a reactor tank breach should occur below the top of the core such that the core also becomes partially or fully uncovered, the core and spare fuel elements would first be moved behind the tank lead shield doors at the opposite end of the tank from the breach location. The movement of spare fuel, in this case, would require the presence of staff personnel in the reactor room. It is possible in this case to supply water to the tank such that a temporary water level above the core could be established. This action would most likely be taken to reduce the exposure levels in the reactor room and thus keep personnel exposures to a minimum during recovery. Actual recovery from such an event would probably involve placing all fuel elements from storage as well as from the core into shipping containers. Enough shipping containers are available on-site to accommodate all of the fuel elements in the core and in storage in a neutronically safe, shielded, and adequately cooled configuration. Estimates of personnel exposures would depend on the actual conditions that existed prior to and during recovery and the fields would be higher than for the previous case. Nevertheless, personnel exposures would be limited during recovery operations by utilizing several members of the staff, limiting their stay times, and providing shielding to the extent possible. It is estimated that individual occupational exposures for such a recovery operation might be on the order of 100-300 mrem, but under no circumstances would an individual be permitted to accumulate more than the limits established under 10 CFR 20.

24. DESCRIBE THE TECHNIQUES AND INSTRUMENTATION USED TO INSURE THE ACCURATE DETERMINATION OF THE REACTOR POWER LEVEL IN BOTH STEADY-STATE AND PULSE MODES. INCLUDE THE CALIBRATION METHODS AND THE FREQUENCY OF CALIBRATION IN THE DESCRIPTION.

Insuring that the available reactor instrumentation correctly monitors the power level in both the steady state and pulse modes is accomplished by performing a thermal power calibration using a calorimetry method. The instrumentation is first calibrated electronically to insure proper linear response. The reactor tank is then isolated from the cooling system and a stirring device is placed in the pool to insure even heat distribution during the calibration. The pool water and tank structure are next allowed to reach thermal equilibrium. Once thermal equilibrium is reached, an electronic thermometer is placed in the pool and the reactor is then brought to an indicated power level of 100 kw. The water temperature is measured at discrete time intervals while at the indicated 100 kw power level. The rate of pool water temperature increase is then calculated using a least squares or other appropriate statistical fit of the data to determine the slope. This slope is then compared to a known reactor tank constant to determine whether the indicated power level is, in fact, the true reactor power level. If the power as indicated is not the same as the true power, the detectors are adjusted to indicate true power. The chamber for monitoring power in the pulse mode is then measured and adjusted if necessary to yield the fractional current at 100 kw such that full pulse power would yield full chamber current prior to chamber saturation. See the figure following for a diagram of the channels. The full calibration is performed annually unless a daily check of the channels indicates that calibration is necessary.



AFRRI CONTROL AND SAFETY SYSTEMS

25. HOW IS THE REACTOR TANK CONSTANT MEASURED?

The reactor tank constant is measured by isolating the reactor pool from the primary cooling and purification system and supplying a known measured source of heat (i.e., an electrical heater) to the pool water. The pool water, tank, tank support structure, core and core support structure act as a heat sink to dissipate the heat generated by this electrical heater. This heat sink is, however, insufficient to dissipate heat above a few kilowatts and therefore the pool water temperature rises at a constant rate in proportion to the amount of heat applied. The specific procedure involves placing a 90 kw electrical heater in the AFRRI pool. The pool temperature is measured at discrete time intervals; the resultant slope (i.e. $^{\circ}C$/kw-hrs) of the pool water temperature is the tank constant. The tank constant has been measured many times; the value has remained constant over the lifetime of the facility. The tank constant has been measured with varying starting temperatures and still found to remain constant.

26. CONSIDER THE OVERHEAD CRANE USED IN THE REGULAR OPERATIONS.

A. HOW OFTEN AND BY WHOM IS IT PROOF LOADED?

B. WHAT IS THE SAFETY FACTOR BETWEEN THE PROOF LOADING AND (1) THE WORKING LOAD RATING? (2) THE MAXIMUM ANTICIPATED LOAD?

The reactor overhead crane is proof loaded annually by the Crane Inspection Department of the Naval Surface Weapon Center. The working load rating is 10,000 pounds. The proof loading is 12,500 pounds. The maximum anticipated load is approximately 4,000 pounds.

27. ARE THERE FORMAL PLANS TO PLACE INSTRUMENTATION IN THE VENTILATION SYSTEM SO THAT DRAW-FAN FAILURES CAN BE READILY DETECTED? IF SO, DESCRIBE; IF NOT, JUSTIFY.

Instrumentation currently installed in the ventilation system to detect draw-fan failures includes a loss of power detector on the motor to exhaust fan number 1 (EF-1). This fan provides primary flow to the main institute stack. The detector actuates an audible and visual alarm in the reactor control room when EF1 draw-fan failures occur. Additionally, draft meters are currently located outside the reactor room, ER1, ER2, the warm storage room, and the hot cell. These meters indicate the barometric pressure differential between the inside and outside of the room and thus provide a measure of the effectiveness of the draw-fan and overall ventilation system. Moreover, a strip chart recorder in the reactor control room indicates and records the flow rate through the AFRRI stack. There are several methods of adding additional alarm functions to the stack flow. One of the methods (loss of flow alarm) has been formally requested by interdepartmental work order and it is expected to be operational shortly.

28. DESCRIBE THE TECHNIQUES AND PROCEDURES USED TO DOCUMENT REACTOR FACILITY CONFIGURATION, AND CHANGES IN CONFIGURATION.

The reactor facility configuration is documented in the individual as-built drawings and is also described more generally in the AFRRI TRIGA MARK-F REACTOR OPERATIONS MANUAL, both of which are updated as necessary. In addition, the reactor Physicist-in-Charge maintains a modification log for planned specific reactor facility minor modifications. The Reactor and Radiation Facility Safety Committee (RRFSC) reviews all proposed modifications. Major modifications also require review and approval by the NRC.

29. ARE THE THERMOCOUPLE CHANNELS THAT ARE USED TO MEASURE THE FUEL ELEMENT TEMPERATURE CALIBRATED? HOW OFTEN ARE THEY CALIBRATED?

The thermocouple channels are routinely calibrated semiannually.

30. UNDER WHAT CONDITIONS IS OPERATION OF THE ^{16}N DIFFUSER SYSTEM REQUIRED?

Operation of the ^{16}N diffuser system is not required under any operating conditions.

31. ARE THE CCTV CAMERAS ROUTINELY USED FOR SURVEILLANCE OF THE EXPOSURE ROOMS BEFORE REACTOR OPERATION? EXPLAIN.

No, the exposure room CCTV cameras are for experimental purposes only, and are not used for pre-operation surveillance. A member of the reactor staff visually inspects the exposure room prior to closing the exposure room plug door.

32. WHAT WATER, SOIL, AND VEGETATION SAMPLES ARE TAKEN FROM THE AREA SURROUNDING THE LIQUID WASTE TANKS?

Environmental samples are collected on a quarterly basis. A water sample is taken from the stream located approximately 200 feet downhill from the storage tanks. Soil samples are gathered from various areas between the storage tanks and the pump house located downhill from the storage tanks. Vegetation samples are collected in the area of the hot waste tank.

33. WHAT MEASUREMENTS ARE MADE TO DETERMINE THAT THE REACTOR AREAS EXHAUSTED INTO THE AFRRI STACK ARE MAINTAINED AT A NEGATIVE PRESSURE? (SEE FIG. 3-5)

Instrumentation currently installed to determine that the reactor areas exhausted into the AFRRI stack are maintained at a negative pressure include draft meters located outside the reactor room, ER1, ER2, the warm storage room, and the hot cell. These meters indicate the barometric pressure differential between the inside and outside of the room and thus provide a direct confirmation of a relative negative pressure in the room.

34. DESCRIBE THE PROCEDURES AND TECHNIQUES USED TO PREVENT AN INADVERTENT RELEASE OF THE CONTENTS OF THE RADIOACTIVE LIQUID WASTE TANKS TO THE PUBLIC SEWER.

All valves associated with the liquid waste tanks are located in a locked pump house. Specific valves that permit discharge of radioactive liquids to the public sewer are individually chained and locked. Keys are kept in a locked box and only the key for the specific tank to be released is taken to the pump house.

35. PROVIDE DETAILS OF THE COMMUNICATION SYSTEM(S) LINKING THE OPERATOR IN THE CONTROL ROOM WITH THE PERSONNEL OPERATING THE ACCESS DOORS TO THE EXPOSURE ROOMS.

The primary communication system is by commercial phone lines between the control room and the prep area. The prep area phone is located next to the exposure room access door control panel. An intercom system provides a separate auxiliary communications channel when required.

36. HOW DO YOU INSURE THAT ANY LEAKAGE FROM THE PRIMARY COOLING WATER CIRCULATING AND DEMINERALIZING SYSTEM GOES INTO THE HOT WATER DRAIN SYSTEM?

Two "cold" drains and one "hot" drain exist in equipment room #2158. One "cold" drain is fitted with a drain stopper which is kept in place except when both the reactor and radiological safety staff are notified and give their approval for its temporary removal. The other "cold" drain is presently being fitted with a cap reducer and ~2 foot high standpipe so that it can accept drainage from non-reactor equipment on a continuous basis without providing a "cold" drainage site for reactor primary water coolant in the event of a leak or spill. The "hot" drain in equipment room #2158 is kept open at all times.

37. HOW ARE THE REACTOR ROOM EXHAUST DAMPERS CHECKED FOR LEAKAGE WHEN THEY ARE CLOSED?

The operation of each reactor room ventilation damper solenoid and its mechanical linkage is checked visually by a member of the reactor staff as part of the normally scheduled monthly maintenance. Checks are also made of the reactor room draft meter when the reactor room is isolated to ensure that a relative negative pressure is maintained.

38. HOW ARE THE "WARM" AND "HOT" LIQUID WASTE TANKS PROTECTED FROM CORROSION? WHO IS RESPONSIBLE FOR THE OPERATION, INSPECTION, AND SERVICE OF THIS SYSTEM? ARE THE WARM WASTE AND HOT WASTE STORAGE TANKS ROUTINELY INSPECTED FOR LEAKS? FOR STRUCTURAL INTEGRITY?

All waste storage tanks and associated piping are protected by an impressed current cathodic protection system. An outside contractor has completed a comprehensive review of the entire system and will make comments on requirements to be included before issuance of a new maintenance contract for periodic inspection and service. The waste storage tanks are inspected weekly for major product loss. In addition, the cathodic protection system is capable of detecting large leaks through a change in the amount of current flow.

39. CONSIDER THE EFFLUENT AIR MONITORING SYSTEMS.

A. DESCRIBE THE AIR PARTICULATE SAMPLING PROBE IN THE REACTOR EFFLUENT LINE.

B. WHAT IS THE FLOW RATE IN THE EFFLUENT LINE AT THIS POINT AND WHAT IS THE SAMPLING RATE?

C. DESCRIBE THE GAS SAMPLING PROBE AT THE TOP OF THE EFFLUENT STACK.

a. The air particulate sampling probe is a plastic tube with a 15/16 inch inside diameter curved downward into the up flowing Phase I air exhaust stack. The diameter of this duct at this point is three feet and the probe is sampling from the center of the duct.

b. The flow rate in the effluent line at this point is typically 12500 cfm which results in a linear flow of 1770 ft/min in the three foot diameter duct. The sampling flow rate is nominally 8.5 cfm through the 15/16" diameter sampling tube resulting in a linear flow rate of 1770 ft/min. This results in reasonable isokinetic sampling.

c. The stack gas sampling probe consists of eight tubes connected at a central point near the top of the AFRRI stack. This results in simultaneously sampling at eight equally spaced points in the AFRRI stack.

d. It should be pointed out that the air released from AFRRI first flows through an absolute filtration system. Thus, AFRRI does not normally release particulate radioactive material. In spite of this and the fact that a particulate monitor is not required by the license, AFRRI has installed and maintained one for more than ten years.

40. ARE ALL OF THE OPERABLE THERMOCOUPLES IN EACH INSTRUMENTED FUEL ELEMENT TIED INTO THE SCRAM LOGIC? DESCRIBE THIS PORTION OF THE SCRAM SYSTEM IN DETAIL.

Not all of the operable thermocouples in each instrumented fuel element are tied into the scram logic.

The thermocouple element found by measurement to be located in the hottest section of the hottest fuel element in the "B" ring of the reactor core is presently routed to fuel temperature safety channel one. The same from the "C" ring is presently routed to fuel temperature safety channel two. The thermocouple output passes a junction (ambient) on the core dolly; from there the voltage enters a fuel temperature amplifier where an output voltage (0-10 volts) is fed to a bistable trip. The bistable trip is set to scram at or below the set point prescribed in the AFRRI technical specifications. The circuit is a scram on fail which prevents operation of the channel (and the reactor) with sections defective or missing.

THE FOLLOWING QUESTIONS ARE BASED ON CHAPTER 6 OF THE 1981 SAFETY ANALYSIS REPORT.

41. P. 6-6. YOU OBTAIN A VALUE OF 0.08 MEV/(CM^3-SEC) FOR S_v. WE BELIEVE THAT IT SHOULD BE 0.18 MEV/(CM^3-SEC). EXPLAIN AND VERIFY THE SUBSEQUENT COMPUTATIONS.

This was a typographical error. The value of S_v should have appeared as 0.18 MeV/(cm^3-sec) as you state. The subsequent calculations using S_v are correct since S_v = 0.18 MeV/(cm^3-sec) was used.

42. P. C-8, TABLE ?. FOR POSTULATED CRITICAL EXCURSIONS THE REGULATORY GUIDES RECOMMEND THE CONSIDERATION OF MORE NUCLIDES THAN YOU CONSIDER IN YOUR POSTULATED ACCIDENT. JUSTIFY YOUR REASONS FOR CONSIDERING ONLY THE NUCLIDES WHICH YOU HAVE.

The answer to this question is available in the response to question #64.

It should, however, be stated here that Kr-83m, Kr-85, Xe-131m, Xe-133m, Xe-133, and I-131 specifically are not included in Table 2, Appendix C because they do not contribute significantly to the activity associated with an assumed 40 MW-sec pulse. This is primarily because these particular isotopes generated in a 40 MW-sec pulse have either relatively long half-lives, low fission yields, or complex decay chains such that their contribution to activity from pulse operation is not significant. It was found that the one-minute decay activity associated with a 40 MW-sec pulse, shown in Table 2, Appendix C, yielded the maximum obtainable activity and that the isotopes listed in Table 2, Appendix C, were the major contributors to this maximum activity, i.e., the other isotopes (namely Kr-83m, Kr-85, Xe-131m, Xe-133m, Xe-133, and I-131) did not contribute significantly to the one-minute decay activity associated with an assumed 40 MW-sec pulse.

43. P. 6-13, PARA. 3. JUSTIFY THE ASSUMPTION THAT 99.8% OF THE IODINE DISSOLVES IN THE WATER?

The value of 99.8% was arbitrarily chosen to provide a conservative estimate since all (i.e., 100%) of the iodine, it is expected, would remain dissolved in the pool water at the temperatures of interest because of the solubility of iodine at these temperatures. General Atomic's experience from aluminum-clad TRIGA fuel element failures, cited on Page 6-11 of the SAR, supports this claim since only noble gases were released. Therefore, the justifiable value here would be 100% and not 99.8%, i.e., 0% would come out of solution and into the reactor room air. However, for conservatism, it was arbitrarily assumed that 99.8% of the iodine remains dissolved in the pool water while 0.2% is assumed to come out of solution and into the reactor room air.

Paragraph 2 under Air and Water Activity Following Cladding Failure on Page 6-13 should be updated to read:

As given in Reference 5, the volume of water in the TRIGA reactor tank is approximately 5.7 E+7 cm^3 and the volume of air in the reactor room is approximately 9.2E+8 cm^3. For the purpose of calculation, it is assumed that all the gaseous fission products in the gap are available for release from the fuel element. As concluded from measurements of the worst cladding failure experienced (Section 6.3.2.1), only noble gas fission products would be expected to escape from the TRIGA pool when the fuel cladding fails. Due to the low pressure in the gap and the small gap size in a TRIGA fuel element, any iodine released from the fuel element due to a cladding failure is expected to be completely dissolved in the pool water. However, for calculational purposes, it is conservatively assumed that as much as 0.2 percent of the

iodine released from the fuel element due to a cladding failure and all of the kryptons and xenons could be released into the reactor room atmosphere. The release assumptions are used to determine the radiological consequences due to the Design Basis Accident of a Fuel Element Cladding Failure (Section 6.3.4.2). Of the 7 curies that could be released from the gap in the cladding failure of an average fuel element, less than 3 curies would be radioiodine while the remaining 4 curies would be krypton and xenon. Therefore, the concentration in the water would...

44. P. 6-16, EQ. 3. CONVERSION FACTORS ARE NEEDED TO YIELD "RAD."

"Rad" here is correct since the conversion factor is built into the 0.25 constant of equation 3 on Page 6-16. Nevertheless, to preclude any misinterpretations, the description of the 0.25 constant value near the top of page 6-17 should be changed as shown below:

0.25 = combined conversion factor and constant for semi-infinite cloud dose (rad-m^3-dis/MeV/Ci/sec)

45. P. 6-17, EQ. 4. IF BR = 3.47 x 10^{-4} m^3/SEC IS THE BREATHING RATE FOR STANDARD MAN, IT SHOULD BE SO STATED.

Your comment is well taken; the description of BR under equation 4 on Page 6-17 should read as follows:

BR = Breathing Rate for Standard Man = 3.47 x 10^{-4} m^3/sec

46. P. 6-17 LAST PARAGRAPH. HOW DID YOU OBTAIN THE ^{16}N ACTIVITY OF 1.2 X 10^{-2} μCi/cm^3? ARE THE UNITS CORRECT?

Question #50 also deals with the N-16 analysis and the actual response to this question can be found in the answer to question #50.

It should be noted here, however, that the number you identify (i.e., the N-16 concentration within the air volume above the surface of the reactor pool, assumed for negligible decay, into which the N-16 activity per second is to be distributed) was determined simply by dividing the maximum activity reaching the pool water surface per second by the volume of air above the pool surface assumed for negligible decay. The units on the N-16 concentration value calculated in this fashion are really μCi/cm^3/sec; however, since we are dealing with a volume for negligible decay into which the N-16 activity per second is distributed, the N-16 concentration value has units of μCi/cm^3 for the assumed air volume of interest.

47. PLEASE PROVIDE A COPY OF THE DOCUMENT STATING THE ADMINISTRATIVE POLICY THAT REQUIRES THAT A FUEL ELEMENT NOT BE REMOVED FROM THE POOL FOR AT LEAST 2 WEEKS FOLLOWING ITS USE IN THE CORE (P. 6-2)

This administrative requirement exists within paragraph 4.k of RSD Instruction 5-8, dated 27 March 1981, which reads:

k. No fuel element which has experienced burnup in the core shall be removed from the reactor pool unless at least two (2) weeks have transpired since its use in the core.

48. P. 6-3, LAST SENTENCE. HAVE THERE BEEN EXPERIMENTS TO CHECK THE 550°C "MAXIMUM AVERAGE" FUEL TEMPERATURE FOR INSERTION OF 2.8% Δ K/K? HOW CAN YOU BE CERTAIN THAT THE FUEL TEMPERATURE WILL NOT EXCEED THE TECHNICAL SPECIFICATIONS LIMITING SAFETY SYSTEM SETTING OF 600°C FOR FUEL TEMPERATURE?

There have been no experiments at AFRRI to confirm the calculated "maximum average" fuel temperature that results from a step reactivity insertion of 2.8% Δk/k (that is, a $4.00 pulse) due to a current license limit of $3.28 (2.3%Δk/k). However, there have been step insertions of $4.00 at other TRIGA reactors. One such experiment made in a 95 element core gave a measured value of 600°C for a $4.00 pulse, i.e., a $3.00 pulse above prompt critical. This temperature was measured in a "P" ring element.

It should be noted that the fuel temperature actually measured during pulsing is not the "maximum average" temperature but usually the hot spot temperature. Approaching the insertion limit in carefully monitored steps (small insertion increases) will insure that the technical specification limits are not exceeded.

49. P. 6-5, EQ. 1. IT IS STATED THAT EQ. 1 GIVES THE ACTIVITY PRODUCTION ASSUMING A SATURATION CONDITION. WE BELIEVE THAT THE EQUATION IS INCORRECT. EXPLAIN THE PRESENCE OF t IN THE NUMERATOR FOLLOWING Σ_a, AND CLARIFY THE MEANING OF "SATURATION CONDITION."

The "t" in the numerator of equation 1 on page 6-5 is in error, i.e., it should not be there. Please delete this "t" from the numerator of equation 1 and from the description of constants and variables immediately below equation 1. Note that the results are correct, however.

In addition to the above deletions, change the one sentence paragraph immediately above equation 1 to read:

Activity production, assuming an Ar^{41} saturation condition has been achieved, can be given as:

50. P. 6-7, PARA. 2 WE BELIEVE THAT THE MACROSCOPIC O^{16} (N,P) N^{16} CROSS SECTION OF OXYGEN IN WATER IS INCORRECT. WE BELIEVE THAT IT IS 6.2 x 10^{-7} cm^{-1}. PLEASE VERIFY THIS AND THE FOLLOWING CALCULATIONS THAT DEPEND ON THE CROSS SECTION. FURTHERMORE, WE REMIND YOU THAT THE REACTION IS NOT PRODUCED BY THERMAL NEUTRONS.

The N^{16} analysis which begins on page 6-7 with paragraph 2 has been redone to correct mistakes such as those you have identified, and should be changed to read:

Another important activation product is radioactive nitrogen (N-16), with a half-life of ~7.14 seconds. As a result of its short half-life, N-16 contributes to occupational exposures of individuals in the reactor room during operation, particularly high power operations, but poses no danger to the health and safety of the general public. The activation occurs as a result of the oxygen content in the pool water from the O^{16} (n,p) N^{16} production process which is exclusively a fast neutron (i.e., ≥10 MeV) induced activation reaction.

Since N-16 is produced by fast neutron activation of oxygen in the pool water within the core region, the activity production and exposure rate equations, cited above as equations (1) and (2), were used to calculate radioactive nitrogen (N-16) releases from the reactor pool to the reactor room air and the associated exposure dose rates that conservatively would be expected above the pool water surface directly over the core.

The concentration of O-16 in water is approximately 0.0554 gm-mole O-16 per ml of H_2O. Using a microscopic (n,p) cross section for O-16 of 1.9 E-29 cm^2 which is averaged over the fission spectrum[5], the macroscopic (n,p) cross section for oxygen-16 in water becomes 6.34 E-7 cm^{-1}.

Using a conservative average fast flux value for the AFRRI TRIGA reactor at 1.0 Mw(t) of 5.0 E+12 neutrons/cm^2/sec, and substituting the appropriate values for ϕ, t_p, and Σ_a (i.e. $\Sigma_{n,p}$ for O-16) into equation 1 yields:

$$A_p = 27.6\ \mu Ci/cm^3,$$

and a total N-16 production rate in the core while at 1.0 Mw(t) of:

$$Q = 0.25\ Ci/sec.$$

Using a measured travel time for N-16 bubbles to rise from the core to the water surface of ~24 seconds, the maximum rate at which activity from N-16 reaches the pool surface is 0.024 Ci/sec.

The radioactive nitrogen will escape from the reactor pool, dissipate, and decay rapidly (~7.14 second half-life) in the reactor room air. The volume of air above the surface of the reactor pool in which the N-16 activity per second is distributed was assumed to be a right circular cylinder with a diameter of 91.5 cm and a height of 100 cm for negligible decay. This assumed volume, therefore, is 6.6 E+5 cm^3. The concentration of N-16 in this volume per second would therefore be 3.64 E-2 $\mu Ci/cm^3$ with a gamma source strength (S_v) equal to 6.2 E+3 MeV/cm^3/sec (E_γ = 4.6 MeV/disintegration). For the N-16 primary gamma photon of 6.13 MeV, the linear attenuation coefficient for air (μ) is 3.2 E-5 cm^{-1} and the flux-to-dose conversion factor (K) is 8.4 E+5 MeV/cm^2/sec per R/hr. Substituting these values into equation 2 for an equivalent volume sphere having a radius (r) of 54 cm results in a conservative estimate of the exposure dose rate due to N-16 at 1.0 MW(t) of 398 mR/hr immediately above the pool water surface directly over the core. It should be noted that the N-16 activity evaluated by this analysis is due only to the amount of N-16 that may be released from the pool. To reiterate, it

should also be noted that due to the short half-life of N-16 and the amount of time required to circulate air from the reactor room to the top of the stack, essentially no N-16 is released from the stack to the environment. As a result, N-16 only presents an occupational exposure potential to individuals in the reactor room near the pool surface during high power operation.

Based upon actual measurements made during reactor operations at 1.0 MW(t), the typical Ar-41 release rate from the pool surface has been approximately 0.5 μCi/sec (September 22, 1980, Ar^{41} Report). Comparing this to the calculated value of 5.1 μCi/sec indicates a factor of ten conservatism in the calculation. A typical Ar-41 concentration at the stack has been approximately 3.0 E-8 $\mu Ci/cm^3$ as compared to the calculated value of 2.7 E-7 $\mu Ci/cm^3$, also indicating a factor of ten conservatism in the calculation.

Gamma radiation levels measured directly over the core just above the pool surface range from 75 mR/hr initially to a maximum of about 200 mR/hr. Gamma radiation levels measured around the reactor pool chain range up to 14 mR/hr. It should be noted that all measured gamma exposure rates (from N-16) include N-16 in both the pool water and the reactor room air. Even though the calculated exposure rate for N-16 of approximately 400 mR/hr only takes into account N-16 in the reactor room air just above the pool surface, it still overestimates the actual dose rate by a factor of two.

51. P. 6-10, LAST SENTENCE. DEFINE THERMAL RATCHETING.

In discussions with personnel at General Atomics, "thermal ratcheting" is a name given to the mechanical stress without full relief which occurs progressively in aluminum cladding and which is induced by thermal cycling. The replacement of aluminum with stainless steel as the cladding material in the standard TRIGA fuel element design has eliminated "thermal ratcheting" as a cladding failure mechanism. This has been verified by testing.

To clarify this in the Safety Analysis Report, the last sentence on page 6-10 should be changed to read:

> Two aluminum cladding failures resulted from mechanical racheting of the cladding material induced by thermal cycling.

The first two paragraphs on page 6-11 should also be changed to read:

> A cladding failure associated with an aluminum-clad element instrumented with internal thermocouples.

In the original TRIGA fuel element design, aluminum cladding was used and mechanical ratcheting occurred during pulsed operations. Modification of the standard TRIGA fuel element design by replacing aluminum with stainless steel as the cladding material during the mid-1960's eliminated this cause of failure. The present stainless steel cladding has greater durability than the original aluminum which experienced the metal cladding failures.

The other cladding failure involving an instrumented element was apparently the result of either internal pressure buildup or water seepage.

52. P.6-2, PARA. 2. PLEASE EXPLAIN THE SIGNIFICANCE OF THE DISTANCE OF "25 METERS FROM THE AFRRI FACILITY." WHAT IS THE SPECIFIC DEFINITION USED IN THIS DOCUMENT FOR THE "AFRRI FACILITY"?

There is no significance to the 25 meters from the facility. This arbitrarily chosen distance simply gives the reader an indication of the dose expected at some arbitrary distance (25 meters) from the AFRRI complex. However, your second question concerning the definition of the "AFRRI complex" brings up a good point. The source of the hypothetical release in this case is the AFRRI stack and not the AFRRI complex. As a result, the text should be changed to read: "35 meters from the AFRRI stack" in both instances in this particular paragraph.

53. P. 6-3, SEC. 6.2.2, PARA 2. UNLESS JUSTIFIED, PLEASE USE CONSISTENT NOMENCLATURE FOR REACTIVITY CHANGES.

β_{eff} for the AFRRI-TRIGA reactor is 0.007 and should have been explicitly stated in the Safety Analysis Report to preclude confusion. β_{eff} will be stated in Chapter 6 and elsewhere where appropriate. Also, whenever a reactivity value for the AFRRI-TRIGA reactor is used in the text, it will be expressed in terms of $\Delta k/k$ as well as dollars ($). However, when reactivity values from other reactor systems are used in the text, only dollar ($) values will be used for direct comparison. This is because β_{eff} for these other systems are not necessarily the same as β_{eff} for the AFRRI reactor.

54. P. 6-3, SEC. 6.2.2, PARA. 3. WE BELIEVE THAT GENERAL DYNAMICS DIVESTED THEMSELVES OF GENERAL ATOMICS BEFORE 1975. PLEASE CHANGE AS APPROPRIATE.

Please change "General Dynamics" in this instance to "General Atomics."

55. P. 6-3, SEC. 6.2.2, PARA. 4. PLEASE CLARIFY THE MEANING OF "A MAXIMUM AVERAGE FUEL TEMPERATURE." IF THIS MEANS THE MAXIMUM TEMPERATURE IN AN AVERAGE FUEL ELEMENT, PLEASE JUSTIFY WHY YOU HAVE NOT EVALUATED THE MAXIMUM FUEL TEMPERATURE AT ANY LOCATION IN THE REACTOR CORE.

Your point is well taken; "maximum average" is not correct here. In fact, it has no meaning since reference here is to the Fuchs-Nordheim model which involves a point reactor with adiabatic heat loss. As a result, the last sentence on page 6-3 should be rewritten to read:

These calculations indicate that a step insertion of 2.8% $\Delta k/k$ that is a $4.00 pulse, (AFRRI-TRIGA technical specification limit) would result in a fuel temperature rise of less than 550° C.

"Maximum average" fuel temperature is a label often given to the expected average fuel temperature in a hot, or maximum, location of a real finite reactor as predicted by the Fuchs-Nordheim model for a point reactor with adiabatic heat loss assumed.

56. P. 6-4, SEC. 6.2.3, PARA. 1. PLEASE CLARIFY THE MEANING OF "SLIGHTLY CRITICAL."

Sentence 3 in this paragraph should be changed to read:

The worst possible case of improper fuel loading would be for an operator to mistakenly insert a fuel element in a core that is already critical at a low power level, i.e. $\leq$1.0 w(t).

57. P. 6-5, SEC. 6.2.4, PARA. 2. WHAT FUEL LOADING IS ASSUMED IN ORDER TO OBTAIN 1 x 10^{13} FOR THE AVERAGE THERMAL NEUTRON FLUX DENSITY? I OBTAIN 0.6 x 10^{13} FOR YOUR CORE. PLEASE EXPLAIN.

If you assume that each fuel element contains a nominal 38.0 gm of U-235, then the AFRRI-TRIGA reactor fuel loading could be either 3.268 Kg U-235 or 3.306 Kg U-235 depending on whether 86 or 87 fuel elements are loaded in the core, respectively. At a power level of 1.0 Mw(t), these fuel loadings correspond to average thermal neutron fluxes of 6.42 x 10^{12} n/cm^2/sec and 6.50 x 10^{12} n/cm^2/sec, respectively

The value quoted in the Safety Analysis Report P. 6-5, Sec. 6.2.4., Para. 2, and elsewhere for the average thermal neutron flux (i.e., 1.0 x 10^{13} n/cm^2/sec) was simply a conservative but fairly representative value for purposes of analysis and calculation.

58. P. 6-5, LAST PARAGRAPH. IT IS STATED THAT THE ASSUMED CONDITIONS REPRESENT "A HIGHLY CONSERVATIVE ASSUMPTION FOR THIS FACILITY." PLEASE EXPLAIN WHY YOU CONSIDER THESE CONDITIONS TO BE "HIGHLY CONSERVATIVE," AND GIVE A QUANTITATIVE ESTIMATE OF THE MAGNITUDE.

Achievement of Ar^{41} equilibrium (i.e., saturated steady-state) conditions within the AFRRI reactor pool water requires operation of the reactor at 1.0 MW(t) for approximately 3 Ar^{41} half-lives or about 5.5 hours. At this point, production rate of Ar^{41} in the pool water equals the loss rate due to Ar^{41} escaping from the pool water to the reactor room air plus decay loss. AFRRI occasionally operates the reactor at 1.0 MW(t) for periods up to one hour, but rarely for longer periods. As a result, you would not expect Ar^{41} evolution rates from the pool water into the reactor room air to be anywhere near those expected under Ar^{41} saturation conditions.

The assumption made in the last paragraph on page 6-5 concerning the 5.1 μCi/sec rate at which Ar^{41} escapes from the pool water is certainly conservative for the AFRRI facility but it is not known without performing actual measurements just how conservative it is. The choice of words here, namely "highly conservative," may have been too zealous and cannot be justified quantitatively without further study and measurements, which are not viewed as warranted in this instance. As a result, "highly conservative" should be changed to read "conservative" which it is known to be in this instance.

59. P. 6-9, FIRST LINE. PLEASE JUSTIFY THE USE OF A DISTANCE OF 25 METERS FROM THE "AFRRI FACILITY."

See answer to Question #52. The phrase "25 meters from the AFRRI facility " here should be changed to read "35 meters from the AFRRI stack."

60. P. 6-9, SEC. 6.3.1, PARA. 1. PLEASE DISCUSS THE RADIOLOGICAL CONSEQUENCES OF THE DEMINERALIZER RESINS COLLECTING ALL OF THE RADIONUCLIDES FROM A FUEL ELEMENT CLADDING FAILURE.

In the event of a fuel element cladding failure, the primary fission products of radiological concern are the radioiodines. It is assumed that failure occurs at saturation for a 1 MW steady state operation. Using a 1.1% gap activity, and a 3-day collection/decay time, the dose rate at one meter would be 5.8 R/hr (Reference Appendix C, Table 1). It should be noted that saturation would not be reached for I-131; therefore, the calculated dose rate is conservative. In addition, the demineralizer could be isolated prior to achieving an unmanageable dose rate, the resins changed, and filtration resumed. At no time would the demineralizer present a hazard to the general public.

61. P. 6-9, SEC. 6.3.1, PARA. 3, LAST LINE. "REACTIVITY" PRESUMABLY REFERS TO CHEMICAL REACTIONS. SINCE "REACTIVITY" IS ALSO A NUCLEAR TERM, IT IS SUGGESTED THAT YOU CLARIFY.

The last sentence on page 6-9 has been changed to read: "The U-ZrH fuel is relatively chemically inactive in water, steam, and air at temperatures up to about 850°C.[2,13]

Reference number 13 has been added on page 6-19 as follows:

13. Simnad, M. T., et al, "Fuel Elements for Pulsed TRIGA Research Reactors," Nuclear Technology, vol. 28, #31, January 1976 pp 31-56.

62. P. 6-10, SEC. 6.3.2.1, LAST SENTENCE. PLEASE CLARIFY THE MEANING OF "PREVIOUSLY USED ALUMINUM CLADDING."

See answer to question #51. In particular, note changes made to the text on pages 6-10 and 6-11.

63. P. 6-11, SEC. 6.3.2.1, LAST SENTENCE. SINCE THE PRIOR OPERATING CONDITIONS WERE NOT GIVEN FOR THE EXAMPLES OF FAILED CLAD FUEL ELEMENTS, IT IS NOT CLEAR THAT THE CONCLUSION "WOULD NOT CONSTITUTE AN UNDUE RISK..." IS WARRANTED UNDER ALL CONDITIONS OF POSSIBLE OPERATIONS. PLEASE JUSTIFY. ARE THESE MEASUREMENTS CONSISTENT WITH THE COMPUTATIONS, PARAGRAPH ONE, PAGE 6-13? DISCUSS.

The previous power history, although important, is not the basis for the conclusion stated. The stated conclusion is based primarily on the fact that only noble gases were released from the failed fuel element and on the rapid 4000-fold reduction in pool water activity after 24 hours. The conclusion also was meant to draw support from the results of the General Atomics experiments mentioned on page 6-10 of the SAR.

General Atomics has provided us with some information about the prior operating conditions for the fuel element which failed and gave the worst-case consequences that are summarized on page 6-11 of the SAR. First, the worst-case clad failure at General Atomics involved an aluminum-clad fuel element that was instrumented with internal thermocouples. This particular fuel element had experienced ~25 Mw-hours of total burnup at the time of failure. Of this 25 Mw-hrs, approximately 3 Mw-hours of burnup were accrued as a result of pulsing operations to as high as $4.00, with the majority of pulses between $3.00 and $4.00. The remaining ~22 Mw-hrs of burnup were accrued through steady-state operations at power levels up to and including 1.4 Mw(t). The actual operating conditions just prior to the cladding failure are not readily available.

The rapid 4000-fold reduction in pool water activity, after 24 hours, measured from the General Atomic cladding failure would correspond to an operating time (at 1 MW) at AFRRI of about 1 hour prior to the cladding failure. This is a typical operating time for the AFRRI-TRIGA reactor. However, using the saturation fission product inventory assumed in the cladding failure analysis for the AFRRI-TRIGA reactor, the pool water activity would experience only a 40-fold reduction in 24 hours. The maximum activity concentration in the AFRRI-TRIGA pool water is calculated to be about a factor of two less than the 0.2 $\mu Ci/cm^3$ reported for the General Atomics cladding failure. However, pool water activity concentration values would be dependent upon the actual water volume involved, the actual amount and type of fission products released to the water, and the extent of mixing in the pool before release of gaseous activity to the reactor room air. As a result, a meaningful comparison cannot be made here directly between the General Atomics failure consequences and the consequences determined by analysis for possible cladding failures in the AFRRI-TRIGA reactor.

The air activity concentration in the reactor room air likewise would be dependent upon the volume of air into which the released activity is actually mixed and the release rate of gaseous activity from the pool water. This also indicates that a meaningful comparison cannot be made directly in this case.

The measured reactor room air activity concentration for the referenced General Atomics cladding failure reached about ten times the MPC_a for fission products. For the cladding failure analysis performed for the AFRRI-TRIGA reactor, the calculated reactor room air activity concentration of noble gas isotopes was about 100 times the MPC_a (submersion) for noble gases (5×10^{-5} $\mu Ci/cm^3$) assuming instantaneous release from the pool water with no holdup or decay in the water considered. The instantaneous dose rate in the reactor room from such an air activity concentration was calculated to be 200 mR/hr.

The reported General Atomics analysis of the activity from the cladding failure collected in the air filters indicates that only noble gases escaped from the pool water. The rapid decay of activity in the reactor pool water indicates that only noble gases were released from the fuel gap at the time of the cladding failure. In the AFRRI-TRIGA reactor cladding failure analysis, nearly fifty percent of the activity released from the fuel gap was assumed to be radioiodines.

Based upon the reported General Atomics consequences for the worst case cladding failure, the conclusion would be that such failures would not constitute an undue risk to the health and safety of the operating personnel or to the general public. Even for the cladding failure conditions evaluated for the AFRRI-TRIGA reactor, which are more conservative, the resulting consequences are not significant and would also not represent an undue risk to the health and safety of the operating staff or to the general public.

The text on page 6-11 obviously must be modified to reflect and clarify these points.

64. P. 6-12. SEC.6.3.2.2, CALCULATED FISSION PRODUCT INVENTORY, PARA. 1, ULTIMATE AND PENULTIMATE SENTENCES; PLEASE EXPLAIN THE SIGNIFICANCE TO THIS COMPUTATION OF THE TERMS "PROMPT FISSION" AND "PROMPT INVENTORIES."

These two particular sentences refer to the fission product inventory expected from an assumed 40 MW-sec pulse operation. Since delayed neutrons play no role during pulse operation, an attempt was made in this paragraph of the text to verbally distinguish the steady-state fission product inventory from that expected from a 40 MW-sec pulse, by use of the words "prompt fission" and "prompt inventory." These designations have no special impact on the computations but rather were used for clarification purposes only. Since these terms have raised confusion, the first paragraph under the section titled Calculated Fission Product Inventory should be changed to read:

Calculated Fission Product Inventory. To determine the various inventories of fission products produced in the core, data were used from reference 7 for infinite steady state power operation at 1.0 MW(t). The resultant full power steady state inventories for kryptons, xenons, and iodines were calculated and are shown in Table 1, Appendix C. The associated fission product inventory for an assumed 40 MW-sec maximum pulse was also calculated using the data from reference 7 with the buildup and decay of activities from Reference 8 considering only prompt fission events in the pulse. The resultant 40 MW-sec pulse inventories for kryptons, xenons, and iodines are shown in Table 2, Appendix C. The activity associated with those isotopes with relatively long half-lives, low fission yields, or complex decay chains is not a significant contribution to the maximum activity present from pulse operations. Such isotopes include Kr-83m, Kr-85, Xe-131m, Xe-133m, Xe-133, and I-131.

These inventories are utilized in the calculation of radiological consequences associated with the various fuel element cladding failures which may result during the operation of the AFRRI-TRIGA reactor. The gamma source strengths were derived using gamma energy decay data from Reference 4 and are shown in Tables 1 and 2, Appendix C.

65. P. 6-15, SEC.6.3.3, LAST PARAGRAPH. PLEASE JUSTIFY USING A DISTANCE OF 20 METERS FROM THE "AFRRI FACILITY."

In this particular case, exposure fields are due to "shine." The initial dose rate for this event was determined to be $\sim$10 mR/hr at the exterior side wall boundaries of the reactor room. "Outside the AFRRI Facility" cited in this

paragraph refers to "outside the east wall of the reactor room" and should be so changed in the SAR here for clarification. The 20 meter distance cited in this paragraph was the distance outside the east wall of the reactor room at which the shine dose rate was determined to be 1 mR/hr at the onset of the event.

66. P. 6-19, REFERENCE 2. IN ADDITION TO THE INTERNAL GENERAL ATOMICS DOCUMENT, THE FOLLOWING REFERENCE TREATS MUCH OF THE SAME INFORMATION, AND IN ADDITION WAS PUBLISHED IN A PUBLICLY AVAILABLE REFEREED JOURNAL: M. T. SIMNAD, F. C. FAUSHEE, AND G. B. WEST, FUEL ELEMENTS FOR PULSED TRIGA RESEARCH REACTORS, NUCLEAR TECHNOLOGY, 28, 31. (1976).

We are aware of this reference and agree with your reasoning. It will be added to the Chapter 6 reference list as reference #13 and cited in the text, where appropriate.

67. P. 6-12, SEC. 6.3.2.2, DETERMINATION OF FISSION PRODUCTS IN GAP, PARA. 2. PLEASE GIVE THE TECHNICAL JUSTIFICATION FOR USING "0.1 PERCENT GAP ACTIVITY" IN THE SUBSEQUENT ANALYSES.

The 0.1% gap activity value cited and utilized for consequence analysis in Chapter 6 of the Safety Analysis Report corresponds to the theoretical maximum value at our fuel temperature technical specification limit of 600°C as well as the expected actual value based on test measurements by General Atomics for the TRIGA fuel temperature safety limit of 1000°C. See references 2 and 13 cited at the end of Chapter 6 for the plot of fission product release fraction as a function of temperature, assuming a cladding failure occurs. Reference 13 is identified in the answer to question #61.

RADIOLOGICAL SAFETY INSTRUCTION (RSI) 220:

PERSONNEL-MONITORING ADMINISTRATION

1. Purpose. This instruction outlines the administration of the Personnel Radiation Exposure-Monitoring Program. The purpose of the program is to:

a. Meet the requirements for compliance with Nuclear Regulatory Commission (NRC) regulations and conditions of the AFRRI/USUHS NRC licenses, and to

b. Provide the means to evaluate any known or suspected radiation exposure.

2. Scope. This program applies to all personnel who enter a restricted area as defined in Title 10, Energy, Code of Federal Regulations: Part 20.3 (10 CFR 20.3). The following areas are restricted areas:

a. All of AFRRI with the exception of the Library (room 3432) and the Conference Room (room 3425).

b. USUHS source usage areas as specified in Experiment Authorizations and posted as "Radiation dosimetry required for all personnel in this area."

3. Regulatory References

a. 10 CFR 19, Notices, Instructions and Reports to Workers; Inspections

b. 10 CFR 20, Standards for Protection Against Radiation

c. NRC licenses R-084 (Reactor), 19-08330-02 (Broad Byproduct Material), and 19-08330-03 (Cobalt Facility).

4. Cancellation. AFRRI Instruction 3000.11

5. Responsibilities

a. The Safety Department implements all aspects of the Personnel Radiation Exposure-Monitoring Program except as otherwise specified.

b. All personnel entering restricted areas are responsible for complying with the provisions of this instruction. In particular, this includes wearing of proper dosimetry at all times and supplying bioassay samples when required.

c. The operator of each large radiation source (i.e., the Reactor, LINAC, Maxitron, Cobalt Facility) and the supervisors of source usage areas are responsible for the implementation and enforcement of the applicable provision of this instruction with regard to the use of radioactive materials in their respective facilities.

6. Implementation procedures to support the requirements of this instruction are described in detail in Radiological Safety Instruction 230 for staff personnel and RSI 240 for supervisors of nonstaff personnel.

7. Exposure limits shall be as set forth in 10 CFR 20 and in applicable NRC licenses. In addition, an operation shall not be planned to exceed an exposure of 50 mrem in one day and/or 100 mrem in one week without prior approval of the Safety Department (SAF).

8. Personnel Dosimetry

a. Devices used to meet monitoring requirements for whole-body, external radiation exposure will be the Harshaw type G-1 and Type NG-67 TLD cards. The card will be worn in a plastic holder positioned at or above the waist. This holder will also contain the individual's security identification credential.

b. Pocket ion-chamber dosimeters may be used as supplementary dosimetry devices and for personnel not requiring dosimetry under the requirements of 10 CFR 20.

c. Other devices such as TLD chips, TLD-impregnated teflon or film will be used as extremity dosimetry during operations in which exposure to the hands exceeds 1.5 rem/hr. An example of operations that may produce such radiation levels are milking radionuclide generators. These devices may also be used for special monitoring procedures. These include monitoring under special environmental conditions, monitoring of special types of sources, etc.

9. External Exposure-Monitoring Requirements. All persons entering the restricted area shall be issued and wear at all times a radiation device as specified below:

a. Persons who have "unescorted" access shall be isued the TLD card-type dosimetry device. A pocket ion-chamber dosimeter may also be required by SAF.

b. Persons who shall be escorted at all times shall be issued a pocket ion-chamber dosimeter.

c. Groups of escorted persons may be provided a single dosimetry device with the prior approval of SAF.

d. Notwithstanding the provisions of 8.b and 8.c, persons whose activities, in SAF's judgment, might result in an exposure exceeding the limits of 10 CFR 10.202 must be issued and must wear dosimetry as specified in paragraph 8.a and/or appropriate supplementary dosimetry.

e. The following restrictions shall apply:

(1) Persons NOT issued an individualized TLD card-type dosimeter shall not have access to and shall not enter any area or room posted as a radiation area, high-radiation area, contamination area, airborne-radioactivity area, or area with a radioactive material sign unless that area has been surveyed and specifically cleared by RadSafe. The radiation facility operator in each individual area is responsible for the enforcement of this requirement.

(2) The dosimeter holder shall normally include a picture security identification credential. Written exception to this requirement may be obtained from SAF if arrangements are made with SAF for special dosimetry, meeting the requirements stated in paragraph 8.a.

(3) Minors (as defined in 10 CFR 20) shall not be issued radiation dosimetry of any kind without prior approval of SAF. Minors who are permitted access to the building shall under no circumstances be allowed in a radiation area, high-radiation area, or airborne-radioactivity area as defined in 10 CFR 20.

10. Internal Exposure-Monitoring Program. SAF will maintain a program for the evaluation of possible exposures due to the uptake and internal disposition of radionuclides. This program will be appropriate for the radionuclides in use at AFRRI and shall meet the requirements of 10 CFR and NRC licenses.

11. Personnel Dosimetry Records

a. Personnel exposure records will be kept in compliance with the requirements of 10 CFR, the Privacy Act, and NRC licenses.

b. All persons on the Personnel Radiation Exposure-Monitoring Program shall be instructed as to the radiation exposure reports that they may request.

c. All dosimetry results shall be reviewed within 10 days of receipt. Head, SAF, shall be notified of any results in excess of 50 mrem for the period.

12. Technical References

a. Guide for Administrative Practices in Radiation Monitoring, Regulatory Guide 8.2, NRC

b. Occupational Radiation Exposure Records Systems, Regulatory Guide 8.7, NRC

c. Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program, Regulatory Guide 8.9, NRC

d. Health Physics Procedures series 1-X, Personnel Monitoring.

RADIOLOGICAL SAFETY INSTRUCTION (RSI) 230:

STAFF PERSONNEL MONITORING

1. Purpose and Scope. This instruction sets forth procedures and requirements to be followed by personnel for whom personnel monitoring is required. It applies to all personnel who enter a restricted area as defined in Title 10, Energy, Code of Federal Regulations, Part 20.3 (10 CFR 20.3). Restricted areas in AFRRI and USUHS are listed in Radiological Safety Instruction 220.

2. Regulatory References

a. 10 CFR 20, Standards for Protection Against Radiation

b. Nuclear Regulatory Commission licenses R-084 (Reactor), 19-08330-02 (Broad Byproduct Material), and 19-08330-03 (Cobalt Facility).

3. Cancellation. None

4. Responsibility. All staff personnel entering restricted areas are responsible for complying with this instruction.

5. Procedures on Initial Check-In. Each new employee will complete AFRRI Form 228 before being issued dosimetry. If an individual indicates on Form 228 that he has been occupationally exposed to ionizing radiation, then former employers (military and civilian) will be contacted in order to secure the individual's radiation history. Until this history is received, the individual will be limited to 1.25 rem per calendar quarter. If it is impossible to obtain an individual's radiation history, an occupational dose will be calculated as required by 10 CFR 10.

6. Procedures for Routine Wearing of Dosimetry. Dosimetry devices will be combined with security devices and will be worn above the waist. Harshaw type G-1 and Type NG-67 TLD cards will be the devices in routine use. TLD cards will be changed periodically on a schedule determined by SAF. The period is normally quarterly but may be monthly or less for certain personnel.

7. Damaged or Lost Personnel Dosimetry Device

a. If a dosimetry device is not returned, SAF will investigate its status. If it has been lost or damaged, an alternative method of dose evaluation will be used, since whole-body doses must be determined for all personnel. The alternative used will be:

(1) Pocket dosimeter, if the individual has worn one during the entire period or for a period in which he was likely to have received over 90% of his total dose.

(2) If no pocket dosimeter is available, a conservative estimate based on co-worker's dosimetry results. If this estimate is non-zero and a backup TLD in the security credential is available, this will then be evaluated.

b. In any case, assigned doses will be reviewed by a senior monitor or health physicist before being entered into official personnel dosimetry records, and the basis for the estimate will be noted.

8. Radiation Exposure Records

a. An individual may request a copy of his/her exposure history at any time.

b. Military personnel will have a copy of their exposure histories made a part of their health records.

c. When employment is terminated, the individual's exposure records will be removed from the active file and placed in an ex-employee folder. These records will be retained by AFRRI in accordance with the provisions of 10 CFR 20.401.

d. Requests from ex-employees for their exposure histories will be answered in accordance with 10 CFR 19.13 (c), provided such requests are accompanied by the signed authorization of the individual to release his records.

9. Personnel Dosimetry Reports

a. All dosimetry readings greater than 200 mrem in one exposure period, or otherwise worthy of special consideration, will be brought to the attention of the department heads/chairmen and Director or Deputy Director of AFRRI.

b. Head, SAHP, will review all personnel dosimetry reports and will send a written report (DF format) to Head, SAF, concerning all notable items. This is to include all dosimetry readings greater than 50 mrem.

c. Reports will be made as required by 10 CFR, Parts 19 and 20.

10. Technical References

a. Occupational Radiation Exposure Records Systems, Regulatory Guide 8.7, NRC

b. Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program, Regulatory Guide 8.9, NRC

c. Harshaw Technical Manuals, TLD system

d. Health Physics Procedures series 1-X, Personnel Monitoring.

RADIOLOGICAL SAFETY INSTRUCTION (RSI) 240:

NONSTAFF PERSONNEL MONITORING

1. Purpose and Scope. The purpose of this instruction is to establish the requirements for those personnel collaborating with or otherwise responsible for outside personnel requiring radiation monitoring.

2. Regulatory References

a. Title 10, Energy, Code of Federal Regulations: Part 19 (10 CFR 19), Notices, Instructions and Reports to Workers; Inspections

b. 10 CFR 20, Standards for Protection Against Radiation.

3. Cancellation. None

4. Responsibilities. Radiation workers who are not employed by AFRRI/USUHS must have a sponsor who is employed by AFRRI/USUHS.

a. The sponsor shall provide a liaison between the nonstaff worker and RadSafe and shall be responsible for the worker's compliance with AFRRI/USUHS radiological safety requirements.

b. For purposes of this instruction, a radiation worker is any individual entering a radiation area with an escort, working with radioactive sources, or otherwise subject to occupational radiation exposure.

c. The sponsor shall provide assistance to the nonstaff worker in required safety paperwork and shall remain familiar with the broad outlines of all radiological work being done by the nonstaff worker.

5. Visitors who, in Safety Department's (SAF's) judgment, may be exposed to radiation exceeding the limits of 10 CFR 20.202 will be issued personal radiation dosimetry by AFRRI/USUHS and will be treated as members of the AFRRI/USUHS staff for personnel dosimetry and record-keeping purposes. Radiological Safety Instruction 230 details procedures for staff personnel. The following two exceptions apply to nonstaff personnel:

a. A history of past radiation exposures will be maintained at the discretion of the SAF professional staff.

b. The individual's employer shall be notified of his exposure at the end of the quarter or at the termination of his work at AFRRI/USUHS, if the individual has exceeded 50 mrem for any one dosimetry period.

6. AFRRI visitors, other than those described above, shall fill out a white information card, AFRRI Form 21, and be issued pocket dosimeters. Dosimeter readings indicating a possible exposure greater than 10 mrem will be investigated, and a report will be made to the Head, SAF.

7. Visitors' exposure records are filed permanently.

8. SAF may waive personnel dosimetry requirements for group visits to the AFRRI/USUHS restricted-access areas, provided the conditions stated in RSI 220 have been met.

HPP 1-6 INTERNAL EXPOSURE MONITORING PROGRAM

1. General.

a. Purpose and Scope: This HPP describes the procedures to be used for bioassay evaluations of radiation dose for AFRRI employees and other personnel potentially receiving occupational radiation exposure from AFRRI sources.

b. Definitions:

(1) Bioassay: The evaluation of radiation dose by radioanalysis of materials which are part of, or have been taken from, the individual's body.

(2) Urinalysis: Bioassay performed by radioanalysis of a urine sample eliminated by the individual.

(3) Whole Body Count: A bioassay by the determination of the activity and identity of principle gamma-emitting radionuclides in the individual's body, by gamma spectroscopy.

c. References:

(1) 10 CFR, Energy, particularly Part 20.

(2) International Commission on Radiological Protection: reports on internal dosimetry, particularly ICRP publications 2 and 10.

(3) Medical Internal Radiation Dose (MIRD) supplements to the Journal of Nuclear Medicine.

(4) NRC Regulatory Guide 8.9, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.

2. Administrative Procedures.

a. Bioassay review:

(1) Each calendar quarter SAHP will review bioassay needs of all groups using or exposed to unsealed radionuclides based primarily upon radionuclide authorizations and radioactive source usage records.

(2) An appropriate quarterly bioassay will normally be required for personnel working with unsealed radioactivity in excess of ten times the quantity listed in appendix C, reference (1) or in excess of Type C lab quantities listed in RSI 5.0 for those radionuclides not listed in appendix C. Adjustment in these action levels may be made by SAHP based on the radionuclide form or special protective measures of particular operations.

(3) An annual whole body count will normally be required for personnel working with unsealed gamma sources in excess of 10 times appendix C quantitites. In addition, whole body counts may be required for certain personnel working with very large sealed gamma sources.

(4) Other bioassay may be required during the quarterly review, but will normally be done as needed on an individual basis.

(5) During dosimetry out-processing for departing personnel a determination shall be made as to whether an exit bioassay is required.

b. Bioassay Procedural Steps:

(1) SAHP will send a DF, "Bioassay Review" (Enclosure 1), to division heads with potential bioassay needs, requesting a return of the enclosed DF, "Bioassay Required" (Enclosure 2), within 7 days.

(2) Based on returns from division heads, SAHP will issue urine sample bottles or schedule whole-body counts, and notify the individuals involved in writing (Enclosure 4.a or 4.b.) and orally. Deadline dates will normally be 3 days from issuance.

(3) Bioassay will be performed and data evaluated.

(4) Pertinent bioassay data will be maintained permanently, on AFRRI Form 119, "Radioanalysis Bioassay Data" (Enclosure 3), by SAHP.

(5) If the individual is on leave or otherwise unavailable, the deadline date will be the first working day after his return. The division head or department head will be informed of the deadline date orally by SAHP.

c. Overdue Bioassay: The following procedure will be followed for overdue bioassays:

(1) The individual will be contacted, 1-2 days after the bioassay deadline. If he is on leave or otherwise unavailable SAHP will contact his division or department head and inform him of the situation.

(2) SAHP will send a DF, "Overdue Bioassay" (Enclosure 5) to the individual through his division head 6-8 days after the due date.

(3) If no satisfactory response is received within 14 days of the deadline date, SAHP will send a DF through SAF to the Radionuclide and X-ray Safety Committee requesting a review of the individual's authorization at the next Committee meeting.

(4) A "Bioassay Status" record will be kept by SAHP on each individual until his bioassay response has been completed. It will include the name of the individual, division, date of sample issue or appointment, date of oral reminder, and date of "Overdue Bioassay" DF.

3. Bioassay Procedures.

a. Urinalysis (URA):

(1) A one milliliter sample will be submitted to the Radioanalysis Lab or SARD, in a ten milliliter scintillation cocktail.

(2) SARD will report all results to SAHP in pCi/ml of specific nuclides. Dose evaluation (Reference 4) will be made by SARD for concentrations approaching legally significant levels.

(3) Radionuclides ordinarily evaluated by URA include: H-3, C-14, P-32, S-35, and I-125.

(4) Unused portions of urine samples will not be disposed of by SAHP until a satisfactory urinalysis data has been received from SARD.

b. Whole Body Count (WBC):

(1) The NIH Nuclear Medicine Group performs WBC upon request. Their phone number is 496-4803. Except for initial and final WBC's, scheduling will be done primarily by SAHP.

(2) Data reported by NIH will be recorded in units of radioactivity (e.g., μCi), never just in counts per minute. Dose estimates, where necessary, will be calculated by SARD in accordance with Reference 4.

(3) Radionuclides ordinarily evaluated by WBC include: Co-60, Sr-85, Mo-99(Tc-99m), Rb-86, and I-131.

c. Other Bioassays

(1) Among the other bioassay methods available at AFRRI on a special request basis are thyroid scans, breath samples, fecal and sweat samples.

(2) Dose evaluations will normally be based on references (2), (3), and (4).

(3) All bioassay results will be retained and be made a part of the individual's permanent dosimetry records.

4. Specific Dose Evaluation Notes.

a. Bioassay Parameters: The following parameters must be evaluated in order to obtain an accurate exposure assessment:

(1) Critical organ and recommended dose limit (legal limit if one exists): This determination may be complicated somewhat by the distinction between the transportable form (ICRP 10) and the soluble form (10 CFR 20B), and estimates of uptake and deposition fractions.

(2) Effective half-life and/or retention time in both the critical organ and the organ or substance to be used for radioassay: This will be crucial in determining both dose commitment and effectiveness of bioassay method.

(3) Energy deposition (including appropriate weighting factor if applicable) in the primary organ of desposition and, for gamma emitting organs. The MIRD (Reference 3) data are recommended, where available.

(4) Time between intake and elimination from the body (if applicable), and time between intake and radioanalysis. Accuracy required in this parameter is a function of effective half-life and/or retention time(s).

b. Sensitivity (low-level detection capability):

(1) Sensitivity will be judged in terms of ability to detect a dose commitment below a certain pre-selected fraction of the maximum permissible dose for the period covered by the bioassay.

(2) When intake date, radionuclide intake form, or other pertinent data are unknown, "worst case" assumptions will be used in evaluating sensitivity capabilities.

(3) Unless otherwise justified and documented, all bioassay radioanalysis will be expected to detect a dose commitment of not more than 25% of the quarterly permissible dose for the critical organ.

WILLIAM R. WEBBER
Acting Head, Radiation Safety Department

DISPOSITION FORM

For use of this form, see AR 340-15, the proponent agency is TAGCEN.

REFERENCE OR OFFICE SYMBOL	SUBJECT
SAF	Bioassay Review

TO Distribution FROM SAHP DATE CMT

1. Please complete the enclosed Form DF, "Bioassay Requirements", and return it to SAHP, Room 1420 by the following date: _____.

2. In order to concentrate bioassay radioanalysis services on those most needing them, and to minimize unneeded bioassays, Radiological Safety requests a report from division heads, each calendar quarter.

3. As a division head, you are responsible for reviewing unsealed radioactive source use by personnel in your division. If no response is received from you, SAF may be required to schedule bioassays for all members of your division.

Distribution:

JOHN M. ARRAS
Head, Operational Health Physics
Division

Enclosure 1: DF Bioassay Requirements

HPP 1-6

DISPOSITION FORM

For use of this form, see AR 340-15, the proponent agency is TAGCEN.

REFERENCE OR OFFICE SYMBOL	SUBJECT
	Bioassay Requirements

TO	FROM	DATE	CMT
SAHP			

1. In response to your request, I have reviewed the radioisotope usage of personnel in this division, on the basis of SAF recommendations in paragraph 2, below, concerning usage during the past three months.

2. SAF will consider bioassay needs for personnel using (a) more than 50% of the amount of radioactivity listed, in any one-day operation, or, (b) as total of more than five times the activity listed, in the previous calendar quarter, or (c) kilocurie amounts of any radionuclide.

Radionuclide:	H-3	C-14	P-32	Cr-51	Tc-99m	I-125	I-131
Microcuries:	100	100	10	100	100	1	1

For information on other radionuclides, contact SAF.

3. The following personnel in this division are recommended for bioassay this calendar quarter, on the basis of the above guidelines:

Name	Radionuclide(amount)	Comments

4. To the best of my knowledge, no other personnel in this division have worked with radioactivity in excess of the amounts listed.

SIGNED: _____

DIVISION: _____ DATE: _____

RADIOANALYSIS BIOASSAY DATA

NAME _____ SSAN _____ GROUP _____

URINALYSIS

DATE COLLECTED	DATE ANALYZED	CONCENTRATION (microcuries/liter) TRITIUM	REMARKS

WHOLE BODY COUNTING

DATE COUNTED	RESULTS: peak energy (keV), cpm; radionuclide, uCi (listed for each peak reported)	REMARKS

ADDITIONAL RESULTS, COMMENTS and REMARKS

a. To be issued with urine sample bottle:

Dear AFRRI Source User:

According to our records you have recently been working with unsealed sources large enough to require a urinalysis bioassay. Please return your urine sample (at least twenty millilite[illegible] to SAHP, Room 1414, by the deadline date shown below. If you have worked with no unsealed sources in the past four months, send this notice to SAHP with a signed statement to that effect. Bioassay is legally required for workers using large amounts of certain radionuclides.

Deadline for sample return: _____, 19__.

b. To be sent as Whole Body Count Schedule:

Dear AFRRI Source User:

According to our records you have recently been working with amounts of gamma-emitting radionuclides large enough to require a whole body count. An appointment at NIH has been scheduled for you on the date shown. If it is not possible to keep this appointment, come to Room 1420 and an alternate time will be arranged. The deadline date is shown below. Bioassay is legally required for workers using large amounts of certain radionuclides.

Scheduled Whole Body Count: _____, 19__, at _____ hours

Deadline for Whole Body Count: _____, 19__.

DISPOSITION FORM

For use of this form, see AR 340-15, the proponent agency is TAGCEN.

REFERENCE OR OFFICE SYMBOL	SUBJECT
SAHP	OVERDUE BIOASSAY

TO DISTRIBUTION FROM SAHP DATE CMT 1

1. On _____ urine samples were requested and whole body counts scheduled for personnel requiring bioassay services this month. As of _____ the following personnel had not submitted their samples and/or had their whole body counts:

NAME	DIVISION	TYPE BIOASSAY

2. Bioassay evaluations are necessary for certain investigators; e.g., those working with significant quantities of unsealed beta-emitters. If bioassays are still overdue by _____, a report will be made to the Radionuclide and X-ray Safety Committee, with a request that they reconsider the use of unsealed sources by persons not supplying bioassays.

3. Please return samples to Room 1414 or 1420, as soon as possible. If you have any questions, contact Mr. Arras, Ext. 50411.

JOHN M. ARRAS
Head, Operational Health Physics
Division

Distribution:

Enclosure 5 to HPP 1-6

RADIATION SOURCES DIVISION
INSTRUCTION NUMBER 5-3

27 March 1981

REACTOR BRANCH EMERGENCY PROCEDURES

1. Purpose. To set forth the emergency procedures to be followed in the event either an auxiliary monitoring system alarmed as a result of a radiation hazard, or a fire developed in the Reactor Building.

2. Applicability. The provisions of this Instruction are applicable to the AFRRI-TRIGA Reactor, the Reactor Branch staff, and the Safety Department staff.

3. Cancellation. RSD Instruction 5-3 "Reactor Branch Emergency Procedures," 14 December 1976, is hereby cancelled.

4. Radiation Hazard. The auxiliary detection systems for the AFRRI-TRIGA Reactor are presented in Table I. In addition the sensor, range alarm setting, and alarm notification are included for each individual system. Any deviation from these established setpoints will require the written authority of the PIC, coordinated through the Head, Radiation Sources Division, and the Head, Radiation Safety Department.

a. Immediate action. The licensed reactor operator will respond to the activated alarms as follows:

(1) If any alarm listed in Table I occurs, the reactor will be scrammed within 15 seconds unless the operator is certain of the cause of the alarm, and that no hazard exists. If the alarm occurred as a result of a malfunction on any instrument required by the Technical Specifications for the AFRRI Reactor (indicated by # in Table I) the reactor will be scrammed immediately. The PIC or his designee will be immediately notified, and the alarm will be recorded in the Reactor Operations Log.

(2) In the event of an alarm from the Stack Particulate Monitor, the operator will ascertain if the LINAC is operating. This information will be considered before determining which action to take in (3) below.

(3) If any alarm listed in Table I occurs, the PIC or his designee will decide which one of the following actions to take:*

*(In the absence of the PIC or his designee, the licensed reactor operator at the reactor console will make the decision himself).

(a) Action #1 - Sound the fire alarm to evacuate the AFRRI complex.

(b) Action #2 - Give oral orders to evacuate the immediate area in the vicinity of the radiation hazard.

(c) Action #3 - No radiation hazard exists and evacuation is not required.

(4) At no time will any reactor operator take actions in response to instrumentation which would be in violation of the Technical Specifications of the AFRRI Facility License R-84.

b. Action #1. The Reactor Branch staff will respond to this action as follows:

(1) AFRRI Instruction 3020.2, "AFRRI Emergency Evacuation and Fire Plan," will be implemented, and the Reactor Branch staff will respond in accordance with that Instruction.

(2) If the reactor is operating, the operator will scram the reactor, insure that all rods are seated, and lock the reactor console. The operator will pickup all the keys in the Reactor Console Room, the Reactor Operations Log and the Emergency Checklist, close the doors to the Reactor Control Room, and then report to the PIC. The operator will turn over to the PIC the keys and the Reactor Operations Log.

(3) The reactor operator will insure that the experimental facilities are secured.

(4) A detailed investigation of the radiation hazard will be conducted, and a written report, summarizing the results of the investigation, will be completed by either the PIC or his designee, and submitted through the Head, Radiation Sources Division to appropriate management levels within 48 hours.

c. Action #2. The Reactor Branch staff will respond to this action as follows:

(1) Same as subparagraph 4b(2).

(2) The Health Physics Division personnel will be immediately notified of the hazardous condition by either the PIC or the reactor operator.

(3) The reactor operator will insure that the experimental facilities are secured.

(4) A Command Post will be established at the nearest safe point to the evacuated area.

(5) If time allows, a two man re-entry team will consist of one member of the Reactor Branch staff, and one member of Health Physics Division staff. The team will suit up accordingly and will enter the evacuated area to determine the type, size, and extent of the radiation hazard. The team will remove injured personnel if required. The corrective action taken will be determined by the information received from the re-entry team. The amount and type of protective equipment carried or worn by the re-entry team will be determined after consideration of the type of hazard existing.

(6) Injured and/or contaminated personnel will be handled in accordance with AFRRI Instruction 6310.1, "Management of Injured and/or Contaminated Personnel."

(7) Same as subparagraph 4b(4).

d. Action #3. The Reactor Branch staff will respond to this action as follows:

(1) A thorough investigation will be conducted by the PIC or senior operator on duty to determine the cause of the alarm.

(2) Corrective action will be taken to insure that the alarm will not sound unnecessarily in the future.

5. Fire Hazard. The Reactor Branch staff will respond to the fire hazard emergency as follows:

a. When the fire alarm is sounded (continuous tone bell), this indicates that AFRRI Instruction 3020.2, "AFRRI Emergency Evacuation and Fire Plan," has been implemented. The Reactor Branch will respond in accordance with the Instruction.

b. Same as subparagraph 4b(2).

c. The reactor operator will insure that the experimental facilities are secured.

d. If the reactor operator on the console discovers the fire he will follow the procedures in subparagraph 4b(2). In addition, he will pull the fire alarm at Door 3162 OR 3154 when exiting the AFRRI complex.

e. Personnel with contaminated clothing will not attempt to change clothes prior to evacuation, but will segregate themselves when outside the building.

f. No attempt will be made to remove fuel elements from the reactor pool during a fire.

g. If the fire occurred in the Reactor facility then an investigation will be initiated as stated in subparagraph 4b(4).

6. Natural Disaster. The Reactor Branch staff will respond to a natural disaster such as flood, tornado, hurricane, earthquake, etc., as follows:

a. If either a forecast is received that a natural disaster is imminent or a natural disaster strikes the AFRRI complex, the reactor operator will immediately scram the reactor, if operating, and then place the reactor in a secured condition.

b. The reactor operator will insure that the experimental facilities are secured.

7. Bomb Threat. The Reactor Branch staff will respond to a bomb threat as follows:

a. Upon receipt of a bomb threat, Administrative Services Division will be notified immediately.

b. The Reactor Division staff will take appropriate action in accordance with AFRRI Instruction 5220.4, "Physical Security Plan for AFRRI", and AFRRI Instruction 5220.4, "Physical Security Plan for AFRRI-TRIGA Reactor Facility."

8. Civil Disorder. The Reactor Branch staff will respond to a civil disorder in accordance with AFRRI Instruction 5220.4, "Physical Security Plan for AFRRI-TRIGA Reactor Facility."

RONALD R. SMOKER, MAJ, USA
Physicist-In-Charge, Reactor
Head, Radiation Sources Division

TABLE I

SYSTEM	TECH SPEC REQUIRED	RANGE	ALARM SETPOINTS	ALARM NOTIFICATION
1. R-1[a]	#	1 to 10^6 mr/hr	a) 500 mr/hr when reactor personnel are on duty b) 20 mr/hr during non-duty hours	a)[1] Local audible buzzer and red light. b)[5] Instrument module - audible buzzer and red light. c)[3] Annunciator Panel - audible horn and red light.
2. R-2[a]	#	1 to 10^5 mr/hr	10 mr/hr*	[5] Instrument module - red light.
3. R-3[a]		1 to 10^5 mr/hr	10 mr/hr*	[5] Instrument module - red light.
4. E-3[a] and E-6[a]	#	1 to 10^5 mr/hr	10 mr/hr*	a)[4][5] Local red light. b)[5] Instrument module - red light.
5. Stack RAM[a]		1 to 10^5 mr/hr	100 mr/hr*	[5] Instrument module - red light.
6. Stack Gas[b]	#	10^2 to 10^6 cpm	10^5 cpm**	[5] Instrument module - red light.
7. Reactor Room Cam[c]	#	50 to $5x10^4$ cpm	10^4 cpm***	a)[1] Local audible bell and red light b)[1][5] Annunciator panel - red light. c)[5] Air damper closed lights on instrument module. d)[5] Instrument module - red light. e) Annunciator panel - audible horn and red light.
8. NMC Criticality[a]	#	1 to 10^3 mR/hr	a) 50 mr/hr - when reactor personnel are on duty **** b) 10 mr/hr - during non-duty hours	[5] Audible bell and red light.
9. Water Box Gamma Monitor[c]		0 to 1.0 ma	0.5 ma	[5] Red light on instrument modul
10. Exhaust Fan	#	n/a	0 CFM	[5] Bell and red light on wall.

a - Scintillation detector
b - Proportional detector
c - GM detector
d - Float activated switch

1 - Reactor Room
2 - Room 3155
3 - Hallway 3101
4 - Prep Area
5 - Control Room

* - Loss of signal alarm: [5]Instrument module - blue light

** - Loss of signal: a) [2,5]Local audible horn and red light; Loss of air pump alarm: [5]Audible horn
b) [5]White light on instrument module

*** - Loss of power alarm: [1]Local yellow light; Loss of signal alarm: [5]White light on instrument module

**** - Loss of signal alarm: [5]White light

ADD TO TABLE I:

SYSTEM	RANGE	ALARM SETPOINTS	ALARM NOTIFICATION
Stack Particulate[c]	10 to 10^5 cpm	2×10^3 cpm	[5]Instrument module-red light