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11 RADIATION PROTECTION AND WASTE MANAGEMENT

11.1 Radiation Protection

NIST has a structured radiation protection program that supports all aspects of NBSR operations. The health physics staff is equipped with sufficient radiation detection equipment to determine, control, and document all occupational radiation exposures. NIST also has established policies that employ the ALARA concept in all operations at NBSR. An environmental monitoring program is also in place to assure that potential radiation exposures in unrestricted areas surrounding the reactor facility are well within regulations and guidelines.

11.1.1 Radiation Sources

In this section, the sources of radiation that are monitored and controlled by the radiation protection and radioactive waste management programs are described. Radiation sources at NBSR can be classified into four general classes:

- Calibration & check sources
- Startup, and other sources used for instrumentation and nuclear support functions
- Airborne, liquid, and solid radiation sources from reactor operations
- Radiation sources produced within the experimental facilities.

The major radionuclide constituents of the radiation requiring monitoring and control by the radiation protection program are summarized below:

Major Sources of Radioactivity		
Airborne	Liquid	Solid
^{41}Ar, ^{3}H	^{3}H, ^{110m}Ag, ^{64}Cu, ^{66}Cu	^{60}Co, ^{55}Fe, ^{59}Fe, ^{65}Zn

Other sources of radioactivity that are found in various reactor and support systems, but are of negligible consequence to occupational or environmental doses are listed below.

NBSR Systems and Radiation Sources (Bolded nuclides are major components)

Primary coolant: **H-3**, **N-16**, Ar-41, Na-24, Mn-54, 56, Cr-51**, Co-60**, Sb-122, 124
Primary pipe (internal contamination): Cr-51, **Co-60**, Zn-65, **H-3**
Helium sweep: **Ar-41**, Kr-85m, 87, 88, Xe-131m, 133,135, 135m, 138, Cs-138
Thermal Shield Cooling System: **Cu-66**, **Cu-64**, **Ag-110m, Zn-65**, N-16
Reactor shield plug/refueling plug: Al and steel activation products, C-14
Air: **Ar-41**, **H-3**, Br-82, Cl-38, Cs-138
CO_2 sweep gas: **Ar-41**, Br-82, Cl-38, S-35
Storage pool: **H-3**, fuel piece cutting products from aluminum activation
Fuel pieces (6061 aluminum, stainless steel): **Fe-55**, **Co-60**, **Zn-65**, Ni-63, Mn-54
Resin beds: **Co-60**, **Zn-65**
Neutron guides: Co-58, **Zn-65**, Ni-59
Pneumatic system: **Co-60, Ag-110m, Zn-65**

11.1.1.1 Calibration, Check, Startup, and Other Radiation Sources

The primary reactor startup source is the combination of irradiated fuel and D_2O, which provide photoneutrons. The backup startup source is an AmBe neutron source that is contained in a solid sealed right-circular cylinder of approximately 1.02” by 1.57” in size (2.5 cm by 4 cm), and has an activity of approximately 2.0 Curies (Ci). It was manufactured by NUMEC of Apollo, Pa., and has a 4.5 MeV average fast neutron emission. It is stored in a shielded container in the reactor source storage room.

Instrumentation check and calibration sources used to support reactor and radiation protection activities are maintained under the Byproduct Materials license (SNM-362) that was issued by the NRC. These include ^{60}Co and ^{137}Cs sealed sources of various strengths (µCi to kCi activities) as well as sealed sources of other radionuclides, and a variety of unsealed sources used primarily for the calibration of laboratory instrumentation (*e.g.*, ^{137}Cs, ^{60}Co, ^{152}Eu, ^{14}C, ^{99}Tc, ^{3}H, and ^{90}Sr). These sources are mostly in the nCi to µCi activity range, in both solid and liquid form. They are stored in Restricted Areas and are subject to periodic surveillance.

The Safety Evaluation Committee reviews fissile and fissionable materials used in experiments that are inserted into the reactor for compliance with the Technical Specifications. These sources are acquired and maintained under NRC License SNM-362. They consist mainly of fission chambers and foils that are used to monitor or calibrate neutron beams and fields. These sources are strictly controlled and periodically inventoried.

Sources produced by the reactor that are related to experimental programs range in activity from aCi (1 attoCurie = 1.0 x 10^{-18} Ci) to kCi. These sources may consist of any chemical element in any physical form. Access to these sources is controlled and they are subject to the radioactive material accountability program.

11.1.1.2 Airborne Radiation Sources

The principal airborne sources of radioactivity associated with the operation of the NBSR are ^{41}Ar and tritium (^{3}H). The only release path for air from the various confinement building ventilation systems is via the building stack exhaust, which has a nominal flow rate of 30,000 cfm (850 m^3/sec). Annual emissions of ^{41}Ar typically ranges from 800 to1200 Ci and ^{3}H ranges from 400 to 800 Ci. This constitutes a dose of less than 2 mrem of exposure to the closest member of the public, which is less than 2% of the NRC dose limit to the public. This analysis was performed with the EPA COMPLY computer code using local wind rose data and computing the dose based on the closest resident in each wind sector, which constitutes conservative analytical boundary conditions.

Monitoring in both the stack and in the building ventilation systems utilizes both installed and periodic sampling. This provides redundant methods for assessing both occupational and public exposure. Occupational exposure is discussed below.

11.1.1.2.1 ^{41}Ar Sources

Argon (^{40}Ar) is about a 0.93% natural constituent of air. Any air volume that is exposed to neutrons will produce ^{41}Ar by the $^{40}Ar(n,\gamma)^{41}Ar$ reaction. ^{41}Ar is a strong beta and gamma emitter with a half-life of 110 minutes. At NBSR, extensive engineering and procedural measures have been taken to minimize ^{41}Ar production. These include:

- Maintaining all heavy water primary systems under positive helium pressure to minimize air in primary water
- Conducting all maintenance activities on primary systems in a way that minimizes air intrusion
- Using cover gases like CO_2 to exclude air wherever practicable in neutron irradiated volumes, such as the cavity around the reactor vessel, and using CO_2 as the driving gas for pneumatic samples being irradiated
- Sealing all penetrations and openings to the extent practicable to exclude air intrusion
- Designing the experiments to minimize neutron irradiated air volumes
- Regularly assessing ^{41}Ar production to verify the effectiveness of the existing reduction measures.

Production of ^{41}Ar at the NBSR is primarily due to the presence of air in the cavity around the reactor vessel. Production associated with experiments is less than 0.1 % of the total because of the smaller irradiated air volumes and because of lower neutron fluences associated with most experiments. The external exposure rate from ^{41}Ar is minimal because the concentrations of ^{41}Ar in the building are less than 1 DAC and the building volume represents a small fraction of a "semi-infinite" cloud. Actual dose rates to a person in the building from a uniform DAC cloud would be less than 0.2 mrem/hr. Personnel dose rates from typical ^{41}Ar levels observed within the NBSR confinement building have been less than 0.004 mrem/hr. This low level, when combined with typical occupancy times and reactor operating frequency results in an annual personnel exposure from this source that is less than 2 mrem. Direct measurements have demonstrated that the calculated values are conservative.

11.1.1.2.2 Tritium

Tritium is produced by the $^{2}H(n,\gamma)^{3}H$ reaction in the heavy water moderator/coolant of the reactor. This produces a primary coolant tritium concentration of 0.3 Ci/liter/yr. As an ALARA measure NIST replaces the heavy water at intervals chosen to limit tritium exposure. All used heavy water is stored onsite until transferred to authorized processors for recycling. With a maximum production concentration of 5 Ci/liter, the radioactivity concentration and exposures discussed below for NBSR would increase by no more than a factor of 5.

During normal operations, the primary release pathway for tritium results from helium leakage into the ventilation system. Since the helium is used to minimize air intrusion into the primary cooling system, it can become saturated with heavy water. Activation of the heavy water produces tritium. Secondary pathways can include various activities, such as refueling or any maintenance activity that exposes heavy water to the air. Abnormal loss conditions, such as a seal failure or a primary coolant boundary failure would be quickly identified by the various monitoring or leak detection systems. The airborne tritium monitoring system at NBSR is capable of detecting a few milliliters of leakage that can occur by water evaporation.

Confinement building tritium levels at a nominal primary concentration of 1 Ci/liter are typically less than 1.0% DAC. Since the operating staff is in the building less than 1500 hrs per year, this represents an annual dose commitment of less than 40 mrem. Bioassay data of the operating staff confirms that most exposures are well below this value. All other personnel are in the confinement building a much smaller fraction of time, and their annual tritium exposures are much less than 1 mrem. Local airborne exposure to heavy water sources by reactor operators during certain activities, such as refueling, can increase their annual exposure from tritium sources, but normally not in excess of 100 mrem.

Abnormal or transient conditions would increase these airborne tritium levels. Previously, when the ventilation system for NBSR was shut down for remediation over a five-day period, the tritium levels slowly approached DAC values. Also, when an auxiliary cooling loop had excessive heavy water leakage, the local airborne tritium levels increased to 1 $\mu Ci/m^3$, which corresponds to 5% DAC.

From a public dose perspective, tritium represents about one-tenth of the dose from ^{41}Ar, assuming equal release activities. Conducting operations in a way that minimizes ^{41}Ar production, even if that results in some increased heavy water loss and minor increases in tritium exposure, results in minimized collective dose because the increased occupational dose to the limited number of operational staff is more than offset by the reduced collective dose to the public. Therefore, ALARA efforts to reduce tritium losses, particularly through ventilation system modifications, must be tempered by possible related increases in ^{41}Ar emissions.

11.1.1.2.3 Fission Products

Noble gas fission products can be detected in the helium sweep system that is maintained over the primary coolant. Those detected radionuclides include gases of Xenon, Krypton, and ^{138}Cs (a daughter product of ^{138}Xe). Using the typical make up rate for the helium system, it is calculated that less than 0.1 Ci of these radionuclides are released annually. These release concentrations are so low (less than 10^{-10} μCi/ml) that they represent a negligible contribution to the total gaseous emissions.

11.1.1.3 Liquid Radiation Sources

The dominant liquid radionuclides of the NBSR are tritium and ^{16}N. Some other minor liquid sources are also discussed in subsequent sections.

11.1.1.3.1 Reactor Primary Coolant

The NBSR primary coolant consists of high purity heavy water. Its primary radionuclides come from the following reactions:

- ^{3}H, produced via $^{2}H(n,\gamma)^{3}H$, a low energy beta emitter
- ^{16}N, produced via ^{16}O (n,p) ^{16}N, a high energy beta and gamma emitter
- ^{24}Na, produced via $^{27}Al(n,\alpha)^{24}Na$, a high energy beta and gamma emitter
- ^{28}Al, produced via $^{27}Al(n,\gamma)^{28}Al$, a high energy beta and gamma emitter
- ^{60}Co, produced via $^{59}Co(n,\gamma)^{60}Co$, a low energy beta and high energy gamma emitter
- ^{51}Cr, produced via $^{50}Cr(n,\gamma)^{51}Cr$, a low energy gamma emitter

Other radionuclides present in the primary coolant include ^{65}Zn, ^{56}Mn, ^{99m}Tc, and ^{122}Sb. These are suspended corrosion products that are activated by neutrons, but are a minor portion of the total liquid radiation source.

During reactor operations, ^{16}N is the primary source of external radiation exposure from the primary piping system. Its short half-life of 7 seconds means that exposure from this source diminishes very rapidly after the reactor is shutdown. At the NBSR, there are two areas where an exposure potential from this source exists, the Process Room and the Monitoring Room. The Process Room contains all of the primary water pumping and processing systems. Dose rates in the process room during 20 MW operation range from a few mrem/hr in relatively shielded or distant zones, to 60 rem/hr in close proximity to the primary reactor piping. A detailed radiation survey of this area for 20 MW power operations is maintained in the event that an entry to the area during operation is required. Entries into the high radiation areas of this room are very rare. If entry to those areas is necessary, the reactor power is reduced whenever possible. The ^{16}N source is present in the Monitoring Room only when the primary sampling system is in operation. General area dose rates when sampling is done at power are less than 5 mrem/hr. Additionally, the time primary water is flowing for sampling purposes is limited by an NBSR ALARA measure, which further limits personnel exposures.

Tritium is produced in the primary D_2O at a rate of 0.3 Ci/kg per year based on 5000 MWd per year. This is equivalent to about 3 Ci/MWd for the 12,000 gallons (45,500 liters) in the primary system. Excluding dilution by makeup, this would eventually result in an equilibrium concentration of about 5.3 Ci/liter. A 1 Ci/liter concentration is used for a number of reference calculations in this document. All used heavy water is stored onsite until transferred to authorized processors for recycle.

As discussed above, tritium is a significant source of exposure from airborne contamination due to evaporation of tritiated heavy water. Exposure by direct contact, thru skin absorption, could result in significant exposures. At a nominal primary coolant concentration of 1 Ci/liter, this represents an exposure of 62 mrem per milliliter of heavy water absorbed. Therefore, any work involving potential exposure by this mechanism requires control measures such as containment, eye protection, gloves, and protective clothing, to minimize and prevent such an occurrence. Individuals that perform this work are required to periodically provide tritium bioassays. The other radionuclides present are at such low concentrations that they represent a negligible residual contamination problem.

^{24}Na is present on the order of 0.1 mCi/liter. It represents a transient external exposure source term in the process room. Due to its short half-life (15 hrs), as an ALARA measure, work in the process room is limited for the first day following shutdown.

^{51}Cr represents the highest activity, longer-lived (half life of 27.7 days) primary system contaminant other than tritium. It is present in the primary at a concentration approaching 0.001 mCi/liter. Since ^{51}Cr emits a low energy gamma, it is almost totally self-shielded by the primary system components. As a contamination source, it is the dominant radionuclide, based on activity, by at least a factor of ten for freshly removed primary components. After several months, ^{65}Zn and ^{60}Co become the dominant residual sources of contamination due to decay of the ^{51}Cr. These radionuclides present an exposure primarily in terms of local external dose due to system contamination. Local "hot spot" radiation sources, at valves, heat exchangers, filters, and resin beds range from a few mrem/hr to 50 rem/hr. Components that have the higher dose rates, such as primary coolant filters and the resin beds, have local shielding to reduce the radiation levels to less than 5 mrem/hr. Exposures from other "hot spots" are controlled through local posting of the areas concerned. The general area dose rate in the process room due to the cumulative effect of these long-term internal contaminants ranges from a few mrem/hr. to about 20 mrem/hr. This room is routinely surveyed, and the survey data is made available for any work performed in this area. In 1995, the measured internal primary surface contamination, after predominately 15 MW power operations, was 0.4 $\mu Ci/cm^2$ for ^{51}Cr, and 0.001 $\mu Ci/cm^2$ for ^{60}Co.

11.1.1.3.2 Reactor Secondary Coolant

The NBSR secondary system consists of plate type heat exchangers and a plume suppression cooling tower that holds about 132,000 gallons (500,000 liters) of light water. Inherent primary to secondary integrity of a plate interface with no seals or welds provides a high degree of confidence that a primary to secondary leak path would not develop. Water sampling of the secondary system has demonstrated the absence of a primary to secondary leakage at a greater than 1 ml of primary water sensitivity in the primary heat exchanges. This is based on liquid scintillation analysis of 10 ml samples of secondary water and the primary system tritium concentration averaging approximately 1.5 Ci/liter. Based upon sampling and the demonstrated level of integrity, there are no likely operational radiological issues associated with the secondary coolant. Nevertheless, this system is subject to regular surveillance.

11.1.1.3.3 Thermal Column D_2O Tank Coolant

The Thermal Column D_2O Tank Coolant is an independent heavy water system. The measured tritium production rate in this system is 25 to 30 mCi/liter per reactor cycle. Equilibrium tritium concentrations in this system typically would not exceed 4 Ci/liter. This is determined by multiplying 30 mCi /liter per cycle times 7 cycles per year times the tritium mean life. Planned periodic replacement of the Thermal Column D_2O will limit the tritium concentration to levels similar to those of the primary coolant water. Losses from this system go to a collection system and the D_2O is recycled. Personnel exposure is minimal from this collection system. Concentrations of other contaminant radionuclides, such as ^{60}Co, ^{51}Cr, and ^{65}Zn, are slightly higher than the primary D_2O. This is due to accumulated contamination in the system prior to its conversion to an independent cooling loop. Since this is prior contamination, the existing levels of 0.5 to 5 nCi/gm should not increase. Additionally, because of the limited volume of this system, these activities are a minor radiological issue for normal maintenance activities, such as resin and filter handling.

11.1.1.3.4 Thermal Shield Cooling System

Cooling of the thermal shield uses purified, light water. The primary radionuclides present in the cooling system include the following:

- ^{16}N, which presents a local dose rate at the ring header ranging from 5 to 30 mrem/hr at one foot, during power operations at various tube locations around the header.
- ^{66}Cu and ^{64}Cu, which are the primary short-term sources of external exposure following reactor shutdown. The ALARA measure for routine maintenance on this system is to delay the start of that work for 36 to 48 hours. ^{64}Cu concentrations up to 1 mCi/liter have been observed.
- ^{65}Zn and ^{110m}Ag, which are the long-term radionuclides in the system. Maintenance procedures are formulated to maximize containment of all fluid transfers to control this potential source of contamination. Concentrations are typically less than 0.01 mCi/liter for ^{65}Zn and 0.001 mCi/liter for ^{110m}Ag.

In the main experimental room, C100, aside from beam experiments, the Thermal Shielding System is the main source of external radiation exposure. General area dose rates are about 0.2 mrem/hr at 6 meters. Local shadow shielding is employed, where practical, to reduce exposures to experimenters working in the room. Control of the cooling water chemistry is also used to minimize the concentration of these radionuclides when practicable.

11.1.1.3.5 Fuel Storage Pool

Water in the spent fuel storage pool is contaminated due to the transfer of spent fuel elements to the pool and from cutting operations performed on them. The major radionuclide present is tritium, at concentrations ranging from 0.01 to 0.2 μCi/ml. Since the volume of the spent fuel pool is 33,000 gallons (124,900 liters), the total pool tritium inventory ranges from 1 to 25 Ci.

Extensive drying of each transferred element is performed as an ALARA measure to limit the amount of transferred tritium. Documentation indicates that about 40 ml of trapped or absorbed D_2O evolves from each element over several months. Since the primary coolant system tritium concentration is approximately 1 Ci/liter, this translates into a tritium release to the pool water of 0.28 Ci per each four-element transfer, or about 2 Ci per year, based on 7 refueling cycles. These small quantities are manageable through simple evaporation or liquid releases.

Cutting of spent fuel elements releases various chips and small particles of aluminum to the pool water. These particles contain the normal activation constituents of aluminum, which are ^{51}Cr, ^{60}Co, and ^{65}Zn. An aggressive spent fuel pool vacuuming program and spent fuel pool filtration maintain these radionuclides to less than nCi/liter levels.

11.1.1.3.6 Miscellaneous Systems

At NBSR, the cold neutron source consists of a liquid/gaseous hydrogen loop cooled by helium. The liquid hydrogen is subjected to a high neutron flux. Consequently tritium is produced in this fluid resulting from the natural occurrence of deuterium in hydrogen. However, this is a closed system designed not to require opening for any kind of maintenance, except for removal of the cold source. Therefore, no operational radiological consequences associated with this system fluid exist.

The Helium Cooling System for the cold source has no exposure to neutrons or to any contaminated system. Therefore, this system has no radiological consequences.

The Liquid Waste System is comprised of selected drains in the laboratory wing and all light water drains from the confinement building. These drains are routed to the liquid waste collection facility. The dominant radionuclide in the collection system is tritium, and is discussed further in Section 11.2.6.

There are no radiological consequences associated with the experiment light water cooling system. Since it is located in the confinement building and contains water, there is the potential for absorption of trace amounts of tritium. However, experience has shown that resultant contamination is negligible.

11.1.1.4 Solid Radiation Sources

Solid sources of radiation at NBSR result from reactor operations. The sources range from very low specific activity, such as used rubber gloves from handling potentially contaminated materials, to intermediate activity items such as activated foils from experiments, and to the high activity spent fuel from the reactor. These sources are described in the following subsections.

11.1.1.4.1 Fuel Elements

All operations involving movement of irradiated reactor fuel elements are performed underwater, which provides the needed shielding. The non-fuel element portions of the spent fuel are removed by underwater cutting and are disposed of separately from the fueled portions of the fuel elements. The radioactivity at the time of shutdown in the pieces from a single fuel element that was used for eight operating cycles is shown in Table 11.1. Only the longer-lived radionuclides are tabulated because of the time delay to shipment. This delay is a minimum of approximately 280 days, but is more typically greater than a year. Table 11.1 shows that when shipments are normally made, the shipment total activity is dominated by ^{55}Fe, ^{60}Co, and ^{65}Zn. For elements used for fewer operating cycles, these values would be reduced, because the neutron exposure time (production time) would be less. Personnel exposure when performing spent fuel handling operations is minimal and is controlled through the use of shielding.

The fission product inventory for one NBSR fuel element is depicted in Table 11-2. Radiation dose rates from these elements are the primary issue for personnel protection. All fuel transfers are performed within a shielded pathway. The room through which the elements are transferred is controlled as a Very High Radiation Area during these transfers per 10CFR20.1602 requirements. All handling of the fuel in the storage pool is monitored with area monitors or survey instruments as a precaution to ensure the fuel element being handled remains adequately shielded.

New NBSR fuel elements nominally contain 350 grams of ^{235}U. Upon receipt they are surveyed for both radiation level and contamination. Prior to insertion into the reactor, each element undergoes a thorough quality assurance evaluation. Dose to operators when handling the new fuel is negligible, since there are no fission or activation products present.

11.1.1.4.2 Reactor Shims

Control shims are the only other high activity component routinely removed from the reactor. This occurs usually every 4 to 5 full-power years. After a minimum decay period of 3 months, the stainless steel hubs are separated from the Cd-Al body and shipped with the other radioactive non-fuel element metal pieces. The Cd-Al shim body is stored in the storage pool or in shielded dry storage wall cavities. Radioactivity for the hub is typically about 20-25 Ci, and the shim body is typically less than 1 Ci. Personnel exposure when performing preparatory operations for shipping is controlled through the use of shielding and the delay between when the shims are removed from the reactor and the preparation of the offsite shipment begins.

11.1.1.4.3 Other Radioactive Solids

Other radioactive solids that contribute to personnel dose and waste volume include:

- Reactor primary resins, which are replaced very infrequently on the order of once every 10 to 20 years
- Reactor primary filters, replaced as needed, usually once or twice a year
- Filters and resins from other systems
- Shielding plugs and related neutron beam shields
- Experiments, or experimental components removed from high neutron flux locations
- Activated experiment samples
- Miscellaneous contaminated materials, such as laboratory waste
- Emergency response.

The radioactivity in these items range from curie quantity material for items such as resins, to barely detectable levels in other items, which constitute the bulk of the waste volume. ^{60}Co in the activated metals, resins, and much of the waste is the primary contributor to personnel external dose rate. This material is stored in restricted areas where access and area dose rates are controlled to limit personnel exposure. Local shielding is used as necessary to limit areas to less than Radiation Area conditions. Sometimes this material is stored in shielded casks or the concrete shield cave facility located in the G-Wing of Building 235. Bulky items with low-level activation, typically experiment shields and components, may be stored in Building 418, which is adjacent to the reactor building. Both of these storage areas are maintained as restricted areas.

11.1.1.4.4 Solid Radioactive Waste Disposition

All radioactive waste is disposed of in accordance with 10CFR20, Subpart K. Solid waste is transferred to organizations specifically authorized or licensed to receive the material, such as the Department of Energy. Materials designated as radioactive waste are transferred to the H wing of the facility for characterization, packaging, and preparation for transfer to authorized recipients. Annual radioactive waste volumes and activities are typically in the range of 126 to 423 ft^3 (11 to 36 m^3) and are less than 1 Ci. In years when unfueled element shipments occur, or major facility modifications performed, larger quantities of radioactive material will be involved. Based on past experience, these are infrequent occurrences on the order of once every 5 or more years.

11.1.1.5 Radiation Sources from Experimental Facilities

NBSR is primarily used for research purposes. The majority of this research involves the use of neutrons to study material constituents, processes, and structure. Therefore, radiation sources will be present in the experimental facilities supporting these activities. These sources are described in the subsections below.

11.1.1.5.1 Neutron Beams

Neutron beams at the NBSR typically range from a few mm^2 to 200 cm^2. Beams with an in-beam dose rate in excess of 100 mrem/hr and accessible (have an open path in excess of 30 cm) are designated as High Radiation Areas. Section 11.1.5.1 has a discussion of beam controls. A characteristic of neutron beams is that the radiation field outside of the beam is typically less than 5 mrem/hr. Sometimes experimental samples or equipment, such as collimators or filters, can result in Radiation Area or possibly High Radiation Area conditions in areas near the beams. These areas are controlled as required by 10CFR20 Sections1601 and 1902. Non-beam related and short-term experiments are shielded and controlled to keep personnel exposures ALARA.

11.1.1.5.2 Thermal Column Facility

This facility is used to provide highly thermalized neutron beams. Although rare, experiments requiring almost the full area of the column, such as for large cross-section exposures involving irregular exposure geometries or full-field exposure geometries, are performed using this facility. Thermal neutron fluxes in the external beams are on the order of $8x10^7$ n/cm^2-sec. This facility is typically controlled as a High Radiation Area, per 10CFR20.1601.

11.1.1.5.3 Pneumatic System and In-core Exposure Facilities

Experiments utilizing these facilities are highly variable, frequently producing multi-curie activity sources. All elements of the activity, facility usage, disposal, and potential personnel exposures are addressed by technical review and administrative authorization processes. Holding the source in a shielded configuration to allow sufficient decay prior to direct manipulation, processing, or analysis is the primary ALARA technique used in these situations.

11.1.1.5.4 Cold Neutron Experiments

The cold neutron guides are fully shielded to the point of neutron beam extraction, wherever possible. At the entry wall to the Guide Hall, the unshielded dose rate from a typical guide is 300 mrem/hr (neutron) and 100 mrem/hr (gamma) at one meter from the guide. The guides vary in size up to 14 in^2 (90 cm^2), and have a cold neutron flux of approximately $5x10^9$ n/cm^2-sec. The in-guide gamma and fast neutron flux rates decrease by the square of the guide length for straight guides. For filtered guides with either bulk or optical filters, the fast neutron and core gamma components of the flux are largely removed. All seven guides in the Guide Hall have primary shutters. These shutters are key controlled, and have status indication (opened or closed). When closed, the design allows unrestricted disassembly and work on experiments for a particular guide.

11.1.2 Radiation Protection Program

In this section, the structure of the organization administering the radiation protection program at NBSR as required by 10CFR 20.1101 is described. The working relationship with other safety and operational organizations is also discussed as well as the authorization basis for the radiation protection program.

11.1.2.1 Radiation Protection Program Staff

Administration of the radiation protection program is performed by the reactor health physics section. This group is administratively separate from the group that manages the operation of the reactor facility. This structure is shown in Figure 11.1.

A Reactor Senior Health Physicist (RSHP) oversees the activities of the Reactor Health Physics Section and is responsible for the implementation of the Radiation Protection Program for the NBSR facility. The RSHP has a separate administrative reporting chain for normal organizational administrative functions, and also for radiation protection program elements related to the NIST materials licenses. Since maintaining the reactor operating license is the responsibility of the Director of the NIST Center for Neutron Research (NCNR), the RSHP has an additional functional reporting relationship to the Director, NCNR. The activities of the Reactor Health Physics Section include:

- Calibration of survey instrumentation
- Effluent and environmental monitoring
- Radiation and contamination surveys
- Personnel monitoring
- Review of proposed experiments and compliance reviews of operating experiments
- Radiological sample analysis
- Training of reactor staff, visiting researchers, and NIST support staff
- Safety Evaluation Committee membership

In addition to the RSHP, the Reactor Health Physics Section is typically staffed with 2 to 4 Health Physicists and 2 to 4 Radiation Protection Technicians. All of the Health Physicists meet the qualification requirements of the Office of Personnel Management for Health Physicists, GS-7 or higher and typically have sufficient training and experience to meet the American Board of Health Physics requirements for comprehensive certification.

Radiation Protection Technicians meet the qualification requirements of the Office of Personnel Management for Physical Science Technicians, GS-5 or higher. They receive additional on-the-job training specific to the NBSR prior to becoming fully qualified as a Reactor Radiation Protection Technician.

11.1.2.2 Plans and Procedures

Plans and procedures for the implementation of the Radiation Protection Program relating to reactor activities may be written by either the operations or the health physics staff. Such plans and procedures, regardless of authorship would be reviewed by appropriate members of both staffs as a minimum, and usually by the Safety Evaluation Committee (SEC) as well. Obtaining final approval from the SEC and document control is assigned to an individual on the operating staff. That person retains the original, signed document as the master and ensures the appropriate distribution to the staff is made. Plans and procedures not directly related to reactor operations, such as instrument calibration, routine shipping and receiving of radioactive materials, are maintained under the NIST materials license and controlled through an analogous structure involving the NIST Radiation Protection Officer and the NIST Ionizing Radiation Review Committee. Plans and procedures that are needed under both programs are either dual approved or are maintained as separate but consistent procedures.

11.1.2.3 Safety Evaluation Committee and Safety Audit Committee

The SEC provides the NCNR with a method for the independent review of the safety aspects of reactor facility operations and health physics, in accordance with Technical Specification 6.2.1 The committee assists the Director of NCNR in examining reactor safety activities, improving the quality of programs, and correcting problems. The SEC is composed of at least four senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology (e.g., nuclear engineering, electrical engineering, mechanical engineering, and radiation protection). If necessary, for particular areas of expertise or review of particular issues, the committee may establish subcommittees using members or outside experts. At least two members are from NCNR and one is from Health Physics.

The Safety Audit Committee (SAC) provides the NCNR with a method for performing independent audits of various aspects of the reactor facility, in accordance with Technical Specification 6.2.2. This committee assists the Director of NCNR in auditing reactor safety activities, improving the quality of programs, and correcting problems. The Director of NCNR appoints the SAC members. The SAC is composed of at least three senior technical personnel who collectively provide a broad spectrum of expertise in reactor technology appropriate to the areas scheduled for audit (e.g., nuclear engineering, electrical engineering, mechanical engineering, and radiation protection) and who are not regular employees of NIST. These two committees are described in further detail in Chapter 12 of this SAR.

11.1.2.4 Interdiction Authority

Any licensed reactor operator, any member of the NCNR management, and any NBSR staff Health Physicist have the authority to interdict and terminate any activity related to the use of the reactor or the use of radioactive materials within the reactor facility that is judged unsafe or that could reasonably lead to an unsafe condition or violation of NRC regulations. Only the licensed operators have jurisdiction over the operation of the reactor itself.

11.1.2.5 Radiation Safety Training Program

To obtain unescorted access to Building 235, individuals must be trained in the following subject areas:

- Basic radiation science
- Meaning and proper response to radiation signs
- Proper use of assigned radiation dosimetry
- Proper response to emergency alarms
- NBSR procedures related to their duties

Individuals who have duties relating to the direct use of radioactive materials or reactor experiments are given additional training, which includes:

- Radiation science specific to their radioactive material usage
- Radiation protection techniques for their specific duties
- Proper use of the appropriate radiation survey instruments
- NBSR procedures and NRC regulations specific to the materials' usage pertaining to usage limitations, ALARA requirements, and material control

Individuals requiring training include the NCNR staff who reside in Building 235, visiting researchers using NBSR experiment facilities, selected plant personnel, and security and fire protection personnel who have unescorted access within Building 235. All trained personnel receive refresher training every 24 months, not to exceed a 30-month interval. Reactor operating staff and the reactor Health Physics staff maintain their radiation safety skills through ongoing training (i.e., on the job training).

11.1.2.6 Records

The following records are retained for the life of the facility, as prescribed in 10CFR20 Sections 2101 through 2110.

- Personnel exposure records
- Radioactive emission determinations and related calculations
- Survey data in areas where radioactive materials are used and any contamination events related to personnel exposure in those areas
- Results of air sampling, surveys, and bioassays required by 10CFR20.1703(a)(3)

All other radiation protection documents (survey records, calibration records, work logs) are retained for at least 10 years. Records relating to 10CFR Part 21 issues are retained for at least 5 years.

11.1.2.7 Part 21 Program

At NBSR, a senior staff member has been designated the responsible individual for receiving, reviewing, and reporting to the NRC any matter relating to requirements under 10CFR21. All staff and facility users required to have radiation protection training also receive training on 10CFR21 reporting requirements. NBSR has promulgated procedures reflecting the 10CFR21.21 requirements. A notice is also posted in a prominent location outlining the reporting requirements and the staff responsibilities required under 10CFR21.6.

11.1.3 ALARA Program

The NBSR ALARA program, as required by 10CFR20.1101, addresses all aspects of NBSR operations. These activities include specific emphasis on proposed new experiments, planned activities involving significant potential personnel exposures, ambient radiation environments within NBSR, and retrospective reviews of occupational and public doses. The Reactor Senior Health Physicist and the SEC have primary responsibility for the prospective analyses, while the SEC and SAC have primary responsibility for retrospective ALARA reviews and audits.

Activities involving the potential for exposures greater than 0.5 person-rem usually require a formal operating plan, a meeting of the involved personnel to discuss the plan, identification of methods for reducing exposures, and specific oversight by the Health Physics staff to ensure that recommended ALARA measures are implemented. Activities involving less potential for personnel exposure have less formal planning and a pre-operational review.

Examples of various ALARA activities implemented at NBSR are identified in Section 11.1.1 above. Engineering controls, such as shielding are utilized to the maximum extent practicable to minimize radiation levels in work areas. Through review of regular surveillance surveys, the Reactor Senior Health Physicist will identify unusual radiation conditions or work practices and recommend improved methods, as well as “lessons learned” feedback to the staff involved.

Minimum ALARA goals are to improve on past performance for ongoing activities and to achieve the lowest exposures by thorough planning. Explicit numerical ALARA goals are rarely established due to the non-routine nature of much of the NBSR research environment. The typical monetary equivalent expended per person-rem avoided is greater than $100,000. The NBSR ALARA goal is to limit radiation doses in unrestricted areas to 10% of 10CFR20.1301(a)(1) using shielding and procedures. An explicit limit for public dose from gaseous effluents has been established at 10 mrem per year, pursuant to 10CFR20.1101(d).

11.1.4 Radiation Monitoring and Surveying

Health Physics supports the NCNR by maintaining portable and fixed monitoring instrumentation as well as laboratory radiological analysis instrumentation. Similar instrumentation located in the Radiation Physics Building provides on-site back up for most of the NBSR instrumentation requirements. Laboratory instrumentation available includes low background proportional counters for alpha-beta counting, liquid scintillation systems for tritium and other low energy beta counting, gross beta counting, and beta spectroscopy, and InGe gamma spectroscopy systems.

Health Physics instruments used for quantitative radiation measurements are calibrated and performance checked at specific frequencies for the radiation measured. Health Physics conducts internal quality control programs to assess the reliability and stability of the laboratory radiological analysis instrumentation. All calibration sources are traceable to either manufacturer supplied reference standards or to national reference standards, such as NIST Standard Reference Materials or NIST primary standards. Radiation instrumentation that provides engineered safety functions related to the operation of the reactor are described elsewhere.

11.1.4.1 Area Radiation and Contamination Monitoring

Contamination and radiation surveys are conducted weekly during operation. During extended shutdowns, alternative schedules are established, which are usually more frequent. Areas that are surveyed include the accessible areas of the Confinement Building and other radioactive material work areas, with emphasis on those areas that pose the greatest potential for changing conditions. These would include the reactor systems and experiments. Additional surveys are performed on an as-needed basis. Typically radiation surveys of active work locations are performed daily.

Spot contamination measurements are routinely performed with paper smears over a 100 cm^2 area. The sensitivity of the instrumentation is typically better than 10 pCi of beta activity and 2 pCi of alpha activity. Large area coverage contamination surveys are performed with floor contamination beta monitors on an as-needed basis.

A full range of portable beta, gamma, and neutron survey instruments are available at NBSR. These include G-M detectors, ion chambers, proportional detectors, plastic scintillation gamma detectors, NaI detectors, and BF_3 and ^{6}LiI moderated neutron instruments. These detectors cover dose rate ranges from 20 μrem/hr to 1,000 rem/hr. Selected portable survey instruments are positioned at various locations around the facility and near experiments for ready use.

Fixed gamma area radiation monitors are positioned at ten selected locations in the confinement building. These ten locations are: three on the C200 level, which includes the control room, top of the reactor, and west wall of C200; four on the C100 level, which includes the experiment and neutron beam room; two in the process room, which contains primary cooling water systems; and one in the spent fuel storage pool area. Alarm set points are specified in NBSR procedures. Typical alarm settings are 5 mrem/hr and adjusted as needed for non-routine activities, generally with the objective of identifying unusual changes in radiation conditions.

Monitors in the spent fuel storage pool area are positioned to detect: increased radiation levels associated with handling of irradiated fuel elements; a loss of shielding from a loss of pool water;

or criticality in the pool. The monitor installed in the new fuel storage area serves as a criticality detector.

Fixed personnel contamination monitors are located at the entrances to the reactor confinement building, and elsewhere on an as-needed basis. Three types of monitors are available for this use. They are:

- Hand and foot monitors using G-M or proportional detectors
- Portal monitors using G-M or plastic scintillation detectors
- Half-body contamination monitors using sealed tube or gas-flow proportional detectors

11.1.4.2 Air Monitoring

Conditions requiring airborne radioactivity monitoring under 10CFR20.1502(b) are rarely present at the NBSR. The two primary airborne radionuclides present at the NBSR are ^{41}Ar and ^{3}H. For ^{41}Ar, area radiation monitors are used to control personnel radiation exposures. Cary ion chambers or gas Marenelli chambers are used to determine airborne activity concentrations, with a sensitivity of better than 0.1 of DAC. An installed gas-flow ion chamber system takes samples from representative areas of the building and from the ventilation system for tritium detection. This system can detect 10% of DAC for tritium levels and is also sensitive to ^{41}Ar. "Cold trap" sampling is also used to sample for tritium. These cold trap samples are analyzed using liquid scintillation, with a sensitivity of better than 10^{-6} DAC.

Continuous air monitors are available for airborne particulate and iodine monitoring on an as-needed basis. One continuous air monitor is typically positioned in the spent fuel storage pool area. Filter and charcoal cartridge samplers are also available for iodine and particulate sampling. These filter and cartridge samples are analyzed in the radioanalysis laboratory.

11.1.4.3 Effluent Monitors

^{41}Ar effluent at NBSR is monitored with a G-M detector located in the stack. This system is calibrated by comparison to a grab sample that is analyzed in the radioanalysis laboratory. The nominal monitor sensitivity is 1.4×10^{8} cpm/(µCi/ml).

The NBSR tritium effluent out the stack is continuously monitored by the building tritium monitoring system. Monthly grab samples from the stack are also collected and analyzed for verification purposes. More frequent sampling or additional continuous monitoring is implemented when unusual or non-routine activities involving the potential for added tritium release are performed. Effluent sampling can also be performed with a particulate filter and charcoal cartridge, and analyzed on an as-required basis.

11.1.4.4 Environmental Monitors

Thermoluminescent dosimeters (TLDs) are used for environmental ambient gamma monitoring. Also used for selected monitoring situations are:

- A pressurized tissue-equivalent ion chamber system with sensitivity of 0.1μrad/hr.
- Environmental G-M monitors with data logging
- A gain stabilized NaI system for monitoring ^{41}Ar or other specific gamma emitters, with a sensitivity of 0.01 μrad/hr for ^{41}Ar

11.1.4.5 Personnel Dosimeters

Personal radiation dosimeters, for both gamma and neutron dose measurement capabilities are provided by a NVLAP certified supplier. Occupational doses can also be determined by pocket ion chambers (PICs) or electronic dosimeters, or by area radiation surveys combined with stay-times. Extremity dosimeters, such as finger TLDs, wrist TLD badges, and wrist PICs are also used when needed.

11.1.5 Radiation Exposure Control and Dosimetry

This section describes how radiation exposure is controlled within the facility and how uncontrolled radioactivity is prevented from entering work areas or the environment. Facility conditions that require protective measures for personnel and the bases for expected dose to workers are also discussed.

11.1.5.1 Exposure Control

The NIST Gaithersburg site on which NBSR is located is a Controlled Area. Shielding, fencing, and controls on radiation sources are applied to limit dose rates in unrestricted areas and doses to the public from direct radiation to values in compliance with 10CFR20.1301. The NBSR ALARA goal is to limit radiation levels in unrestricted areas to 10% of 10CFR20.1301(a)(1) using shielding and procedures.

Building 235, the NCNR building, is under security access control that requires a key or appropriately coded ID card for entry, and is controlled as a Restricted Area. Other areas, such as the south yard and Building 418, may be established as restricted areas on an as-needed basis. Aside from engineered controls, the primary exposure controls are appropriate training and aggressive surveillance oversight. Unescorted access to the facility requires appropriate training (see Section 11.1.2.5). Dose rates in office and generally accessible work areas are limited to 10CFR20.1301 levels. Radioactive material usage is limited to designated areas within the facility.

Many of the areas around beam experiments and around the local shielding for beam experiments are posted as Radiation Areas. Radiation levels around experiments are minimized by careful attention to:

- The materials placed in neutron beams to minimize prompt gamma radiation, such as the use of cadmium and gadolinium as mask materials.
- Designing the experiment for minimum exposure to surroundings and personnel.
- Performing an ALARA review and use of shielding where appropriate.

The design requirement for long-term experiment shielding is less than 5 mrem/hr at one foot from the shielding for routinely accessed areas, with an ALARA goal of less than 0.5 mrem/hr at that distance. Dose rates of up to 100 mrem/hr at local "hot spots" may be permitted where added shielding is not practical, but only if the impact on personnel exposures is minimal. Dose rates of up to 100 mrem/hr may also be permitted in areas not routinely accessed, such as areas between guide shields. In all cases 10CFR20.1902 controls are applied.

Neutron beams with an in-beam dose rate in excess of 100 mrem/hr., (the High Radiation criterion), that are accessible (have an open path in excess of 30 cm) are controlled as follows:

- Rooms containing such beams are posted with a message indicating that the neutron beams are High Radiation Areas
- The point at the shield from which the beam projects is posted with the sign "Caution - Neutron Beam"
- A barrier or structure, sometimes the experiment itself, will be present to prevent inadvertent entry to the beam area
- A beam stop will be added to limit the path length to the minimum needed for the experiment.
- The experiment will be equipped with a local shutter and a local shutter control to stop the neutron beam when not needed.
- A prominent visual indicator will be present to alert persons in the vicinity of the neutron beam when the shutter is open.
- A detector capable of sensing entry to the beam area will be present to provide both an audible and visual local warning signal, whenever the beam is on.

Large area neutron fields in excess of 100 mrem/hr are controlled in accordance with 10CFR20.1601. Neutron beams with absorbed dose rates in excess of 500 rad/hr will be controlled in accordance with 10CFR20.1602. At present, there are no neutron beams in excess of 500 rad/hr operational at the NBSR. Long-term High Radiation Areas within the facility, such as the process room, are controlled by permanent internal procedures and comply with 10CFR20.1601.

Total annual exposure for the staff at NBSR over the last 10 years has ranged from 7.4 to 8.7 person-rem for 500 to 800 workers. This excludes the few years that involved high-exposure maintenance and major upgrade activities, where the yearly exposure ranged from 18 to 22 person-rem.

11.1.5.2 Personnel Dosimetry

Only a few persons at the NBSR facility, typically less than 20, meet the 10CFR20 Subpart F requirement for personal monitoring. In all cases, workers are monitored in the manner required by 10CFR20.1502. The NBSR policy for issuance of personal radiation exposure monitors is much more conservative. Typically all the persons assigned to the NBSR have a personal dosimeter. Temporary personnel requiring monitoring under the 10CFR20.1502 requirements, such as visitors, will normally be issued pocket ion chambers (PICs) or a NVLAP badge.

Individuals that could exceed 500 mrem in a year are also issued a pocket ion chamber for administrative trending of their exposure, which can be read on a daily basis. These individuals also have their radiation badges processed at least quarterly and their pocket dosimeters read at least monthly. Minors are usually not permitted to work in areas where their duties could result in an annual exposure in excess of 100 mrem. In the event that such work is permitted, those individuals will be monitored as required by 10CFR20.1502.

The average researcher annual exposure is less than 20 mrem, and the maximum staff annual exposures rarely exceed 500 mrem during routine operations. Potential exposures that exceed regulatory limits to special populations, such as embryos or declared pregnant women, are very limited. In these rare cases, added surveillance is provided and work is adjusted to further limit exposure to radiation and to radioactive materials.

Monitoring for internal exposures from airborne radioactivity as per 10CFR20.1502(b) is rarely required at the NBSR. Nevertheless, selected personnel are monitored for tritium exposure as an ALARA and quality assurance measure. Additionally, a whole body counting facility is available for assessing potential internal exposures from gamma emitting radionuclides. Annual reports are made to the NRC as required by 10CFR20.2206(a)(1). The use of Planned Special Exposures is not envisioned at the NBSR. In the event that such are needed in the future, a program will be established as required by 10CFR20.1206.

11.1.5.3 Respiratory Protection Program

NIST does not use a respiratory protection program to limit exposure to radioactive material as described in 10CFR20.1703. NIST does use an OSHA related respiratory protection program and respiratory protective devices at the NBSR for non-radiological protection purposes, such as pipe welding and cutting. Sometimes this work is performed in areas with a radiological involvement. However, in these cases, engineering controls are instituted in a manner that protection from airborne radioactive material under 10CFR20.1703 is not required or needed.

11.1.5.4 Radioactive Material Control

Sources produced by the reactor related to experimental programs range from aCi to kCi in activity. They can be in all physical forms and can involve virtually any chemical element. For beam experiments, the activity levels are typically less than 1 μCi. All experiments are reviewed with regard to the production, use, manipulation, control, and disposal of radioactive material. Appropriate controls related to exposure minimization, such as shielding and security, are established for the anticipated activities and radionuclides during the experiment approval process. This approval process might include special administrative controls, a walk-through of planned activities, or even a test irradiation at reduced conditions with less materials involved and a reduced neutron flux. Sources possessed under NRC License SNM-362, but used during the conduct of any reactor experiment, will be similarly reviewed and controlled.

Radioactive material is transferred from the reactor experiment areas only to approved usage areas. All materials removed from the reactor experiment areas are surveyed prior to removal, particularly for contamination. This is to prevent inadvertent transfer of radioactive materials. Any radioactive material to be transferred from the NCNR facility is required to have prior Health Physics clearance.

11.1.6 Contamination Control

The primary contamination control measure is to control sources of potential contamination such that the spread of contamination is avoided, preferably with engineered controls, such as negative pressure ventilation and enclosed handling in hoods or glove boxes. At NBSR any removable activity found is cleaned to non-detectable levels (less than 0.2 dpm/cm^2 of removable beta activity over 100 cm^2). Reasonable decontamination efforts are made to achieve the NBSR goal of no detectable removable activity being present in any area. In areas where external exposure is high and benefits of decontamination efforts minor, minimal efforts are performed for ALARA reasons. An example of this would be the process room, because the related external exposure for decontamination efforts is greatly in excess of the potential internal exposure.

The NBSR criteria for establishing contamination controls for an area and for equipment and items to be released for unrestricted use are based on the NRC Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, July 1982, (re-issued April 1993). Reasonable decontamination efforts shall be made to achieve the goal of no detectable removable activity being present on any items or equipment released to an Unrestricted Area. Items with detectable radioactivity, either removable or fixed, for which decontamination is not possible or practical, will be retained for radioactive material disposal when identified as unneeded.

11.1.7 Environmental Monitoring

The NBSR Environmental Monitoring Program is designed to verify that the radiation doses to the public are less than 10CFR20.1301 requirements. The methods used involve active effluent sampling and monitoring, performing environmental surveys, and monitoring of liquid waste releases. These methods are described in paragraph 11.1.4.3. Real-time monitoring instruments displayed in the reactor control room are capable of recognizing a potential elevated release from the NBSR, since normal operational releases represent a negligible fraction of the regulatory limits. Reviews of the recorded release data are also performed quarterly. Public dose is based on measured emissions and is determined by computational models. At NBSR, the EPA COMPLY code is an example of several that are used for these computations. Continuous, passive radiation monitoring at the NIST site boundary using TLDs or similar monitoring devices is also performed. The individual device dose sensitivity is less than 1 mrem, but the sensitivity for detecting an annual dose above background is a factor of ten higher.

Environmental surveys include radiation surveys, sampling of grass and soil, and water sampling of local streams and ponds. The collected samples are analyzed for possible activation radionuclides and fission products. Water samples are also assayed for tritium. Environmental samples of water, soil, and grass are collected and analyzed at least quarterly from a minimum of four locations for each sample type. Soil samples are collected during the non-growing season (October through March), and grass samples are collected during the normal growing season (April through September). This analysis typically has a sensitivity of better than 1 pCi per sample. Liquid scintillation analysis of water samples is also done and typically has sensitivity better than 10 pCi/ml.

11.2 Radioactive Waste Management

In this section the overall radioactive waste management program for NBSR is described. At NBSR, the radioactive waste management program is assigned to the health physics group. One health physicist within the group, under the supervision of the Reactor Senior Health Physicist, is assigned primary oversight for this program. Waste management is an integral part of the Radiation Protection Program and is subject to all the management and oversight measures of that program. The operation of the NBSR and the experimental facilities is conducted to minimize radioactive waste production consistent with ALARA objectives. Disposition of gaseous effluent is discussed in sections 11.1.1.2 and 11.1.4.3.

11.2.1 Solid Radioactive Waste Controls

Solid radioactive waste is defined at the NBSR as any item that has been identified as having no further usefulness, and is contaminated with detectable radioactivity, either removable or fixed, or can reasonably be presumed to be contaminated based on its usage history and knowledge of the related processes, and further decontamination is not practicable.

Radioactive waste is segregated from non-radioactive waste. This segregation is primarily based on process knowledge, e.g. knowing where the material was used or from what system the material originated. Items that are exposed to neutrons or to sources of contamination are considered potentially radioactive. These items would include irradiated portions of experiments or items that came in contact with primary reactor coolant. Sometimes the usage knowledge provides a reasonable presumption that the item can be decontaminated. If decontamination of an item is successful, as determined by a radiation survey and contamination check, only then will that material be released for unrestricted use or disposal.

Characterization of solid radioactive waste is done by direct assay, involving sampling and direct gamma spectroscopy, as well as by process knowledge. The following are examples of process knowledge application:

- For neutron-activated materials, activation calculations are performed using the full knowledge of the constituents of the material irradiated. This technique is applied to identify radioactive constituents that are undetectable by routine survey techniques.
- For materials in liquid systems, the knowledge of the radionuclides present in those systems is applied. In particular, if the material were exposed to primary water, the tritium analysis would be based on the maximum moisture content possible in the material and the concentration of tritium in the primary at the time the material was exposed.

Determination of whether a material is radioactive or not is usually made through the survey process. Instrumentation in combination with process knowledge is used to determine both the activity level and constituents of any radioisotopes present in the material. These surveys are conducted in low background radioactivity conditions and explicit criteria for a positive indication above background is established. An example of this criterion is 5 μR/hr for a 1" x 1" (2.5cm x 2.5cm) NaI, 'μR' survey instrument. The instrumentation used at NBSR includes:

- NaI detectors using rate or scalar mode counting for gamma emitters
- Shielded plastic scintillation detectors, mainly for determining contamination on tools
- G-M or windowed proportional detectors for beta emitters

Solid radioactive waste is accumulated at the point of production and collected consistent with keeping exposures ALARA. All accumulation containers are appropriately labeled. Collected low-level waste is typically transferred to the H wing of Building 235. Records of the origin of the waste and its radiological contents are kept in preparation for packaging and shipment. Other waste requiring special handling or containing high levels of radioactivity, such as primary filters and large neutron beam shields, are stored at other locations.

Systems, components, and experiments are designed so as to prevent the production of mixed waste (toxic and radioactive) to the maximum extent practicable. Any such waste (i.e. lead or cadmium) exposed to neutrons, is segregated and stored until disposal at an authorized facility is arranged.

All solid radioactive waste is disposed of by either transfer to licensed disposal sites or processing facilities. It is transported as required by 10CFR Parts 61, and 71, and by the applicable state licenses of the recipient. Detailed radioactive waste characterization documents and manifests are prepared and retained in accordance with 10CFR20.2006.

11.2.2 Solid Waste Minimization

Since the costs of solid radioactive waste disposal are high, materials with low activation potential are used wherever practical to minimize the production of radioactive waste. At NBSR, experiments are designed to be reusable and to minimize the amount of neutron activated material, and the amount of materials used in processes that become contaminated are minimized to the greatest extent practicable.

Disassembling and segregating also minimizes radioactive waste. In this way only the contaminated or activated portions of objects are disposed as radioactive waste. To the extent practicable, a commercial, HEPA-filtered compactor is used to minimize the volume of radwaste. It is typically used for laboratory paper waste and contaminated gloves.

11.2.3 Gaseous Waste

The three gaseous waste streams of the reactor facility are the Normal Air, Irradiated Air, and Process Room ventilation systems. The radionuclides present in these systems are discussed in Section 11.1.1.2. Processes that might generate airborne or gaseous contamination are vented through one of these systems. Gases in these systems are passed through HEPA filters prior to release up the stack. For an upset or abnormal operating condition, these ventilation systems go into a recirculation mode of operation, and a standby charcoal filter is made operational.

11.2.4 Liquid Waste

Selected drains in the laboratory wing and all light water drains from the confinement building are routed to the liquid waste collection facility. This facility consists of a 1,000 gallon (3,785 liter) tank, two 5,000 gallon (18,900 liter) tanks, various filters, and related pumps and valves. Water collected is sampled and analyzed for its radioactive constituents and then filtered to meet 10CFR20.2003 solubility requirements prior to release to the sanitary sewer. Credit is taken for the daily NIST site release volume of approximately 260,000 gallons (984,100 liters) to meet the 10CFR20.2003 concentration limits.

If unanticipated quantities of radioactive material are accumulated in the system, the capability is present to either circulate the water through filters or resin beds to reduce the radionuclide concentration, to transfer the water to containers for off-site processing at a NRC licensed facility, or to store it and allow radioactive decay to reduce the level of activity. As an ALARA measure, the general operating practice, when practicable at the NBSR, is to collect any high activity liquid waste at the source and separately process and dispose of that waste.

The dominant radionuclide at NBSR is tritium, with annual releases on the order of 2 to 5 Ci. Tritium releases will comply with the limits contained in 10CFR20.2003(a)(2) and (3). Annual releases of other prominent beta-gamma (^{60}Co, ^{65}Zn, ^{110m}Ag) emitters are typically in the range of 0.1 to 1 mCi, with an average concentration of less than $3x10^{-9}$ µCi/ml. Releases to the sanitary sewer under NIST materials license SNM-362 are a small fraction of the total NBSR liquid radioeffluent produced. These releases are included in the totals for compliance with 10CFR20.2003(a)(2) and (3). Confinement Building air conditioning condensate is the major contributor to the liquid waste volume. This is due to low-level tritium contamination in the building air. The annual volume of radioeffluent waste released is typically 300,000 gallons (1,135,500 liters), which is diluted by the NIST site sanitary sewer volume of approximately 100 million gallons (378.5 million liters).

11.2.5 Long Term Storage

The policy at the NBSR is to dispose of items identified as waste. When that cannot be done expeditiously, there is a long-term storage area for radioactive waste material located in the G wing of Building 235. At this location there are 33 shielded concrete cavities, each about 10 feet (3 meters) deep and varying diameter. This is the primary location for storing items that present a significant exposure potential, but have potential future use. It is also used to store some items to allow radioactive decay to reduce the activity level prior to disposal.

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Table 11.1: Long-Lived Isotopes in the Unfueled Portion of a Spent Fuel Element

Radionuclide	Half life	Activity (Ci)
Fe-55	**2.7 y**	**3.3**
Fe-59	44.5 d	0.45
Co-60	**5.27 y**	**0.7**
Zn-65	**244 d**	**2.6**
Mn-54	312 d	0.3
Cr-51	27.7 d	0.0033
Ni-63	100 y	0.016
Ni-59	76000 y	0.00013
Co-58	70.9 d	0.37
Hf-181	42.4 d	0.063
Sc-46	83.8 d	0.0068
Sb-124	60.2 d	0.0012

Table 11.2: Fission Product Inventory for one Fuel Element

Nuclide	Activity (Ci.)	Nuclide	Activity (Ci.)	Nuclide	Activity (Ci.)
KR 85	7.91E+01	SN123	2.10E+01	BA137m	6.16E+02
SR 89	2.47E+04	SN125	5.50E+01	BA140	2.65E+04
SR 90	6.29E+02	SB125	3.42E+01	LA140	3.01E+04
Y 90	6.76E+02	SB127	3.27E+02	CE141	2.90E+04
Y 91	3.05E+04	TE127	4.04E+02	CE143	2.68E+03
ZR 95	3.26E+04	TE127m	9.20E+01	PR143	2.75E+04
NB 95	3.29E+04	TE129	3.19E+02	CE144	1.57E+04
NB 95m	2.37E+02	TE129m	4.91E+02	PR144	1.57E+04
ZR 97	2.37E+02	TE131m	1.27E+02	PR144m	1.88E+02
NB 97	2.38E+02	I131	1.06E+04	ND147	9.21E+03
NB 97m	2.25E+02	XE131m	1.69E+02	PM147	1.25E+03
MO 99	9.54E+03	TE132	8.19E+03	PM148	3.13E+03
TC 99m	9.19E+03	I132	8.44E+03	PM148m	2.03E+02
RU103	1.60E+04	I133	7.01E+02	PM149	2.65E+03
RH103m	1.44E+04	XE133	2.29E+04	PM151	1.24E+02
RH105	3.75E+02	XE133m	3.61E+02	SM153	1.25E+03
RU106	9.73E+02	CS134	6.82E+02	EU154	3.08E+01
RH106	9.73E+02	CS136	1.53E+02	EU155	1.73E+01
AG111	7.94E+01	CS137	6.51E+02	EU156	2.23E+03
Total Activity (Ci) = 3.97E+05					

Data Source: ORIGEN Code Output performed by BNL on May 7, 2003.

Note: The Fission Product Activity listed here is for one NBSR fuel element that has been irradiated for 304 days (8 cycles) at an average power of 667 kW and “cooled” for 5 days in the core prior to removal (NBSR administrative policy).

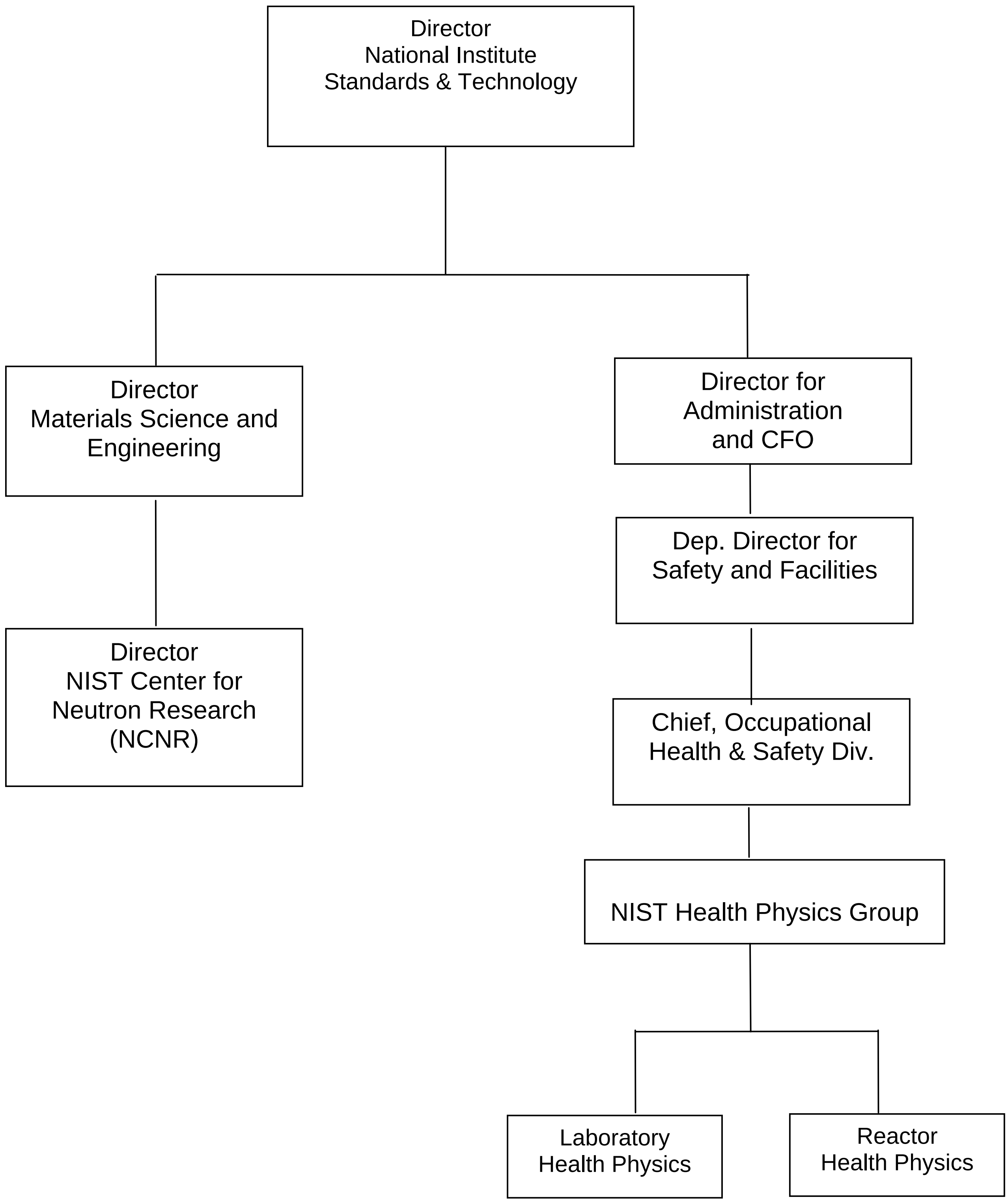

Figure 11.1: NIST Organizational Chart