RADIONUCLIDE CHARACTERIZATION OF SPENT PURIFICATION FILTERS

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ABSTRACT

A remote handling facility is described which will aid in the assessment of the radionuclide mix and total activity content of spent purification filters. The facility will also dry filters as part of their preparation for off-site shipment. One possible system design, consisting of a robot with five degrees of freedom, a shielded Ge-Li detector, and a vacuum drying system, is described. Benefits could include reduced operating and waste storage or burial costs, feedback on filter performance, and an improved understanding of environmental protection requirements.

INTRODUCTION

In conventional (nonradioactive) fluid systems, filtration is required to remove particles which could interfere with the operation of equipment. In nuclear power stations, filters serve the additional function of removing radioactive corrosion and fission products from the fluid; this reduces the radiation fields on system equipment, and reduces radioactive contamination in areas where leakage of reactor fluid occurs.

A filter design frequently used in reactor applications is a cylindrical cartridge, with flow directed from the outside of the cylinder to the inside. The filter material is normally resin-impregnated paper, whose surface area is maximized by a pleated arrangement. The pore size is often less than 1 micron, in order to trap the large fraction of radioactive particulates which are in this size range.

A practical filter assembly will consist of several cartridges housed within a steel pressure vessel (Figure 1). During use, the fluid pressure drop across the filter assembly gradually increases due to particulate accumulation on the cartridges. When the pressure drop reaches a predetermined value (often identified by the filter manufacturer), either the individual cartridges or the entire filter assembly is changed.

Various degrees of mechanization have been tried to facilitate the filter changing process [1,2,3]; due to the substantial radiation fields involved, special precautions are required in all phases of processing this waste:

(1) Water removal,

(2) characterization of waste type,

(3) packaging,

(4) shipping, and

(5) disposal or indefinite storage.

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In general nuclear stations could carry out the first three of these operations in a more orderly and efficient manner with improved resources. In this paper, a facility is outlined that would address this need. As an illustration, the operation and benefits of the system are described for a 4-unit CANDU-PHWR station. For perspective, a review of current filter handling systems (and associated problems) is first presented, before describing the proposed facility.

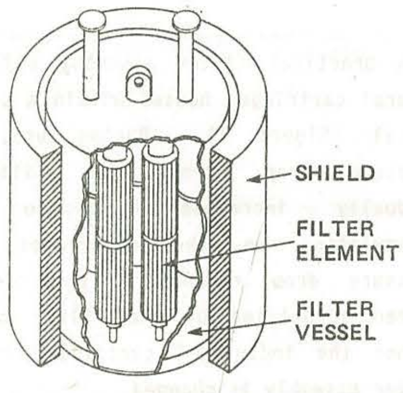

FIGURE 1: MULTI-ELEMENT CARTRIDGE FILTER

REVIEW OF CURRENT SPENT FILTER HANDLING SYSTEMS

Virtually all fluid systems in a nuclear reactor, eg, primary heat transport (PHT), moderator, and gland seal continuously cycle the water through a purification circuit consisting of ion exchange columns and/or filters. To change a spent filter, it must first be valved off from the system, drained, and then dried. The latter task is normally accomplished by forcing air through the filter and venting the exhaust to active waste treatment.

After drying, the filter is removed from the reactor system for measurement of activity level, packaging, and shipment. At Pickering Nuclear Generating Station-A (PNGS-A), the removal operation is performed using a remotely operated hoist. The operator manipulates the hoist by watching its movements through a mirror, while transferring the filter into a heavily shielded shipping flask (Figure 1). These manoeuvers have been eliminated in the PNGS-B design by allowing the filter to reside in its own shielded flask from the initial in-service point up to final implacement at a storage site. At both PNGS-A and PNGS-B, a crude estimate of filter activity level is made based on the measured doserate on the outside of the shipping flask.

Neither station, however, has a system that can meet two important criteria: complete drying of a spent filter, and accurate characterization of the waste as to radionuclide type and activity level. The significance of these two points merits a brief explanation.

(1) Water Removal

Tests conducted within Ontario Hydro have shown that approximately 0.5 kg of water could remain trapped on each feed-and-bleed purification filter element after draining. (Annual filter usage from a 4-unit CANDU station is summarized in Table 1.) The potential consequences are as follows:

(a) Small quantities of water may be trapped in a filter which is shipped to the waste storage or disposal site. The presence of liquids in waste is generally considered to be undesirable.

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(b) Water trapped in filters can cause handling problems for station personnel.

(c) For spent CANDU filters, the estimated commercial value of the trapped D_2O is on the order of $500 to $1000 per reactor per year, based on Table 1.

TABLE 1

DOSE RATE DISTRIBUTION ON FILTERS PRODUCED ANNUALLY BY BRUCE NGS-A AND PICKERING NGS-A

Dose Rate	Bruce NGS-A Purification* System Filter Elements	Pickering NGS-A Purification* System Filter Elements
< 1 rem/hr	5	13
> 1 rem/hr < 15 rem/hr	35	11
> 15 rem/hr	95	13

*Includes filter elements from the PHT Reactor, Gland and Fueling Machine Purification Systems.

(d) The presence of contaminated water may further complicate the task of estimating radioactivity concentrations on the filter.

(2) Waste Characterization

It is impossible to accurately determine the type and activity concentration of radionuclides in a filter when it is packaged in a vessel with 5 inches of lead shielding in its walls. Therefore, the characterization of radionuclide particulates on filter elements has historically been based on laboratory studies of particulate filtered from small quantities of reactor liquids. As shown by Table 2, the tests show several radionuclides contained in the water, but it is not known what quantity of each is trapped by filters (as opposed to remaining in the fluid system, or being removed via IX columns). Also, such tables give no indication as to the variability of radionuclide mix among filters.

TABLE 2

ACTIVITIES OF FISSION AND CORROSION PRODUCT RADIONUCLIDES IN PARTICULATE FILTERED FROM 1 LITRE OF "TYPICAL" PHT D_2O*

Radionuclide	n Ci/litre of D_2O (Average of 10 Samples)
Zr/Nb-95	900
Ru-103	820
Ru-106	420
Cs-137	24
Ce-141	50
Ce-144	750
Co-60	2
Zn-65	6
Sb-124	3
Sn-113	2
Cr-51	5
Ba-140	5
Cs-134	11
Sb-125	7
Ag-110M	4

*Based on the analysis of samples from Bruce NGS-A, Unit #1.

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Similarly, efforts to estimate the total field strength are frustrated by the thick shielding of the shipping flask. Correlations between the field strengths inside and outside of the flask are strongly dependent on the radionuclide mix, which is unknown. Efforts to manually place a gamma survey meter inside the flask to obtain a direct reading are awkward, and could result in exposure to personnel.

In summary, the following consequences are believed to result from waste characterization problems:

(a) It is difficult to conform to regulations which require the radionuclide content of waste to be accurately characterized. New US regulations (10 CFR61), which specify several radionuclides that must be individually measured, cannot be met by conventional techniques.

(b) Environmental studies are hampered by a lack of data on radionuclide source. As a result, waste transportation, storage, and disposal facilities are designed to accommodate the uncertainty in hazard level. This leads to a significant economic penalty.

(c) A decay profile for the activity of each filter cannot be established. Such information would allow each filter to be managed as economically as possible. For example, a filter could be transferred from its temporary heavily-shielded storage area to a less expensive long-term facility after a calculated decay time.

(d) Filter design for nuclear application could be optimized by knowing the concentration and spatial distribution of radionuclides on the spent filters.

PROPOSED FILTER HANDLING SYSTEM

(1) Outline

A filter handling facility is proposed that would strive to meet the following criteria:

(a) Safety: Operations that would result in a significant radiation dose to personnel, or awkward manipulations with equipment not designed for the task at hand should be minimized.

(b) Fluid Removal: The filters should be completely dried before shipment from the station.

(c) Waste Characterization: Standards specified by the US Code of Federal Regulations, Part 10, Chapter 61, should be met.

The proposed facility is shown in Figure 2. The remote handling equipment consists of a "robot" (mobile crane) with 5 degrees of freedom: 3-dimensional translation, rotation about the vertical axis, and an on-off filter grabbing mechanism, or "gripper". A video camera is included to facilitate remote control of the handling mechanism. Apart from the rotational movement and a requirement for computer-assisted control, this system is similar to that being installed at the

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Darlington Nuclear Generating Station (DNGS). The vacuum drying system, and computerized waste characterization equipment indicated in Figure 2, however, are not included in the Darlington design.

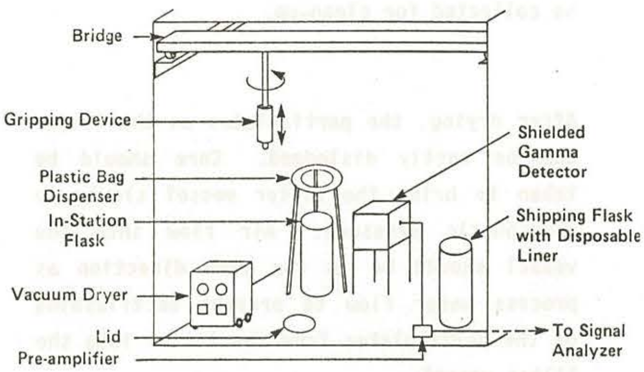

FIGURE 2: PROPOSED FILTER HANDLING FACILITY

(2) System Operation

The following sequence of operations to process a spent filter is proposed; minor modifications would be required to suit the peculiarities of each station's purification system design:

(a) A spent filter is valved out of service, drained, and air-blown dry to the extent possible.

(b) A vacuum drying system, described in Subsection 3, is connected to the filter to complete the drying operation.

(c) A mechanical gripper is remotely latched on to the spent filter.

(d) The filter is hoisted out of its shielded in-service location and placed in a temporary shielding flask. A new filter is then installed in the purification circuit. This task could be operator-controlled using the video camera.

(e) The disposable filter vessel (or individual cartridges in the case of a reusable vessel) is lifted out of the flask by the crane, carried over to the radionuclide characterization equipment, analyzed, and then placed in a disposable container located inside a shipping flask. Details of the analysis procedures and equipment are described in Subsection 4.

(f) A crane places a lid on the shipping flask, which is then manually secured.

(3) Filter Drying

The drying system can be a portable installation or fixed if all filters are dried at one location. The conventional method is to pass air through the filter for a period of about 24 hours. The water displaced from the filter can be collected by an air drying system. This method employs existing station facilities (instrument air and active exhaust), but has the disadvantage of only removing 75 to 85% of water from the filter. Further drying of the filter is required to provide consistent filter radioactivity characterization measurements and to prevent contamination of the work area. This additional drying can be achieved by a vacuum system employed as a backup to, or independently of, the air drying system.

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A vacuum drying system consists of vacuum hoses, vacuum gauges, relief and isolation valves, a vacuum pump, and water vapour collection and handling facilities. A configuration for a vacuum drying system is shown in Figure 3. The filter vessel to be evacuated is connected via a valving and vacuum gauge arrangement to a vacuum surge tank. The surge tank provides for a more regulated evacuation of the filter assembly and protects the vacuum pump from slugs of water which can occur if the filter vessel is not drained or if water condenses in the vacuum line. The surge tank will require a drain valve and should have the same capacity as the largest filter vessel. A typical filter vessel of 0.2 m^3 capacity could contain 4 kg of water after draining.

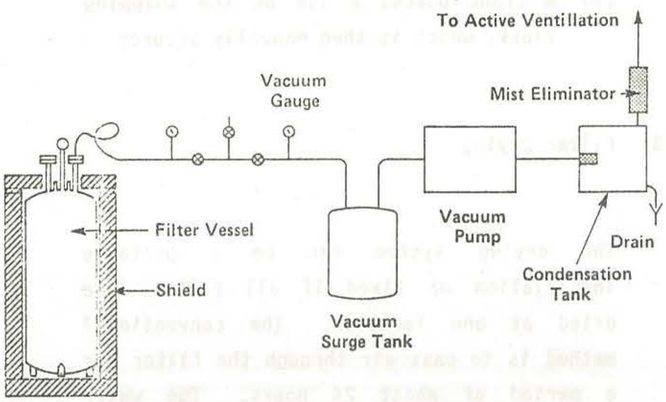

FIGURE 3: VACUUM SYSTEM CONFIGURATION

Various vacuum pump arrangements could be used to evacuate the filter vessel. A suitable pump for this application is a two-stage vacuum blower with a canned motor. These pumps, which are recommended for toxic or radioactive gas applications, can achieve pressures in the range of 0.15 Pa (10^{-3} Torr) at pumping speeds of 5.5 x 10^4 cc/s, (100 cfm).

A condensor following the vacuum pump will extract water from the vacuum pump discharge. This water can be drained to a container, piped to a cleanup area, or released to a drainage sump. For filters used in heavy water systems, the water will be collected for clean-up.

After drying, the particulates on the filter can be easily dislodged. Care should be taken to bring the filter vessel slowly to atmospheric pressure. Air flow into the vessel should be in the same direction as process water flow to prevent backflushing of the particulates from the filter into the filter vessel.

(4) Waste Characterization

The objective of characterization is to determine the concentration of each prominent radionuclide on the filter as a function of position. For multi-element filters sealed in disposable vessels, this will require that activity measurements be made as a function of axial position Z and circumferential position θ. In the case of disposable cartridge elements inside a reusable vessel, the elements are sufficiently thin (approximately 4" diameter for PHT filters), that there would be no significant θ-dependence of radiation fields, and therefore axial measurements will suffice.

The sequence for radionuclide analysis would be as follows:

(a) The robot would grip a filter cartridge (or disposable filter vessel) and lift it out of the temporary storage flask. To prevent contamination spread, this manoeuvre would be used to pull a

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disposable polyethylene bag off a retaining ring located above the vessel. The bag would elastically self-seal at top and bottom (Figure 2). Where the process water flow is from the inside to the outside of the filter, it may be possible to eliminate the polyethylene bag.

(b) The robot would take the element to a heavily-shielded Ge-Li detector with a single columnar opening in its shielding wall. A preamplifier would be contained with the detector inside its shielded cavity; the remainder of the analysis equipment (amplifier, multi-channel analyzer, etc) would be located remotely (Figure 4). A small computer would be essential to process the large quantity of data.

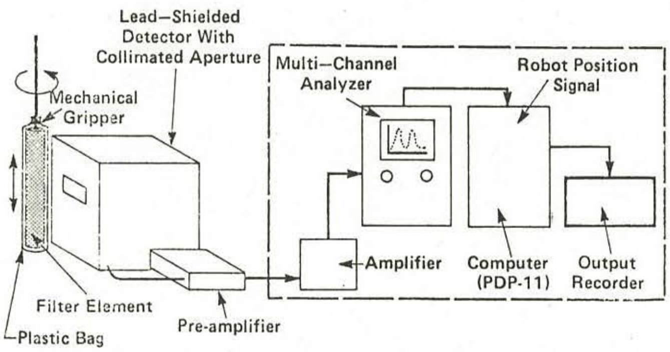

FIGURE 4: RADIONUCLIDE CHARACTERIZATION EQUIPMENT

(c) The robot would position one end of the filter at a preprogrammed position near the opening to the shielding wall, and pause while the activity is counted. Note that some form of automatic positioning mechanism is required, as the activity count rate would be very sensitive to small errors in positioning. Computer control of the facility for these manoeuvers is therefore desirable.

(d) A permanent record would be made of the count rates at energies corresponding to prominent particulate radionuclides.

(e) The robot would then move on to allow activity measurements at a sequence of preprogrammed positions.

(f) A minicomputer would use these data, combined with a signal indicating the filter's position, to construct an activity profile of the filter element. From these results, a decay profile plus doserate-versus-time chart can be constructed, and an appropriate selection made as to shipping and storage/burial facilities.

(g) The filter element would be placed in a disposable container inside the shipping flask, and the operation repeated for the next element.

An obstacle which prevents the design of an economical filter handling facility is the large variety of filter designs used within a single station. Some degree of standardization is strongly recommended. To reduce waste storage/burial costs, the exclusive use of disposable filter cartridges (with reusable vessels) would be an important step in this direction.

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SUMMARY AND CONCLUSIONS

The proposed filter handling facility would require a substantial financial investment for design and construction, particularly if existing stations were to be backfitted. However, such a move would bring the following benefits:

(1) The facility would meet the requirements of 10 CFR61 in characterizing the radionuclide content of waste.

(2) The vacuum drying system would remove the traces of water trapped in the filters sent for storage or disposal, and promote additional recovery of D_2O for heavy water reactors.

(3) Because the characteristics of each waste item would be accurately known, the conservatism of current storage and disposal system designs might be relaxed; this could lead to a considerable financial saving, while a more accurate filter classification system would promote equivalent or improved standards of environmental protection.

(4) Knowledge gained as to the spatial distribution of various radionuclides in a spent filter could lead to improvements in filter design or mode of operation.

(5) Worker safety would be enhanced by the presence of a completely automated filter handling system.

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