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ABSTRACT

Ontario Hydro operates a 10,000 MW capacity nuclear power system utilizing the CANDU pressurized heavy water reactor design. The use of D_2O as moderator and coolant results in the production of about 2400 Ci* of tritium per MWe-yr. As a result, there is significant Canadian experience in the treatment, handling, transport and storage of tritiated wastes. Ontario Hydro operates its own reactor waste storage site which includes systems for volume reduction, immobilization and packaging of wastes. In addition, a facility to remove tritium from heavy water is presently being commissioned at the Darlington nuclear site. This facility will generate tritiated liquid and solid waste that will have to be properly conditioned prior to storage or disposal. The nature of these various wastes and the processes/packaging required to meet storage/disposal criteria are judged to have relevance to investigations in fusion facility waste arisings. Experience to date, planned operational procedures and ongoing R&D in this area are described.

1.0 Tritiated Waste Management

Ontario Hydro operates its own reactor waste transportation system as well as its own waste storage site which includes systems for volume reduction, packaging, and waste storage. Most of the solid reactor waste contain absorbed tritiated moisture, Reference 1.

1.1 Reactor Wastes Characteristics

A majority of the tritiated waste is miscellaneous low level solid waste arising from operation and maintenance of the CANDU reactors. A large proportion of this waste is discarded protective clothing, cleaning materials, piping, valves, tools and other hardware.

* 1 Ci = 3.7×10^{10} Bq

Waste segregation into combustible, compactible, and non-processible categories is carried out at the stations. In preparation for transportation to the waste site in 1 m^3 Type A metal containers, the the waste is collected and packaged and sealed in clear polyethlyene bags.

In addition to the miscellaneous wastes, tritiated heavy water purification media such as filters and ion exchange resins require storage. Spent filters are dewatered, sealed in metal containers and shipped to the waste site in Type B transportation flasks. Most of the spent ion exchange resin is handled in bulk and is effectively detritiated by dedeuteration (water washing) and dewatering and shipped to the storage site in 3 m^3 carbon steel cylinders.

Only a small volume of low specific activity tritiated liquid waste is presently generated by Ontario Hydro which is diluted to very low levels before release to the environment. However, the startup of a heavy water detritiation facility at the Darlington Nuclear Generating Station site has caused Ontario Hydro to consider, in detail, the tritiated wastes that will be generated and the handling systems etc., required. These wastes include some highly tritiated liquids and solids which will require conditioning prior to storage.

Typical annual volumes and tritium concentration of wastes processed and stored at the waste operations site are summarized in Table 1. A simplified schematic of the solid waste management system is given in Figure 1.

TABLE 1

TRITIATED REACTOR WASTES CHARACTERISTICS

Type of Waste	Annual Volume m^3	Annual Tritium Activity Ci
Combustible	3500	1050 – 10,500[a]
Compactible	1100	1650 – 8250[b]
Non-Processible	900	1300 – 5400
Disposable IX Resin Vessels	12	5[c]
Filters	10	50 – 100[c]

a) Calculated based on Incinerator Stack Emissions
b) Waste Sampling Data ranged from < 4 mCi/m^3 to 125 Ci/m^3
c) Estimated Value

1.2   Waste Processing

The waste operations site consists of a waste volume reduction facility,
(WVRF), which is equipped with an incinerator, baler, compactor, and
storage facilities. As significant volumes of tritiated liquid waste
requiring solidification are not produced in CANDU reactors, an
immobilization system is not normally required.

Combustible waste (less than 60 mR/hr*), is volume reduced (50:1) by
incineration and, in doing so, the tritium is eliminated from the waste.
No attempt is made to recover the tritium released from the waste during
incineration, and therefore it is discharged directly to the stack.   The
tritium free incinerator ash is unloaded into 2.5 m^3 sealed metal
containers which are placed in engineered storage structures.

Compactable waste (less than 200 mR/hr) is volume reduced by mechanical
compaction. As with the incinerator, no attempt is made to recover
tritium released into exhaust ventilation systems during the baling or
compacting processes. The plastic bags containing the waste are
compacted into 0.4 m^3 cardboard boxes which are then secured with steel
straps and double plastic bagged.

1.3   Waste Storage Facilities

Various types of storage facilities are used to provide multiple
confinement envelopes between the waste materials and the subsurface
environment. Storage is considered interim, i.e. waste which does not
decay during the lifetime of the facility will be retrieved and sent to
disposal. All wastes stored are a solid form, retrievable and placed in
engineered storage structures with a design life ranging between 50 and
100 years.

Storage structures have been constructed on a dense clay till deposit.
The clay till provides a natural confinement barrier in which low
permeability (10^{-6} to 10^{-7} cm/s), and geochemical retardation
ensures that any radioactivity escape from the engineered storage
facility and under-drainage system will move very slowly. The types of
storage structures are:

Concrete Trenches

Shallow inground reinforced structures, 3 m wide x 3 m deep x 40.3 m
long, used for low level waste. When full, they are sealed with 0.31 m
thick concrete lids using neoprene gaskets.

* 1 Rem/h = 10 mSv/h

Concrete Tile Holes

Inground vertical cylindrical structures, 0.69 m internal diameter x
3.5 m deep are used for ion exchange columns and filter vessels. They
are sealed by backfilling with a high slump concrete.

Quadricells

An above ground concrete structure, 6 m square x 5.5 m high divided into
four cells used for bulk spent ion exchange resin (virtually tritium
free). They may also be used for high level reactor core components.
Concrete caps are used to seal the cells.

Storage Building

An above ground concrete structure, the low level waste storage building
(LLSB), is 50 m long x 30 m wide x 8 m high, used primarily for
incinerator ash containers and baled waste. It is equipped with fire
extinguishing, lighting, smoke detection, ventilation and drainage
systems. All waste is pre-packaged in self stacking containers.

1.4   Operational Experience

A large volume of radioactive waste containing approximately 35000 Ci of
tritium has been handled and is being stored at the site. Occupational
dose has been low, approximately 6 to 12 Man-Rem per year of which about
5% is due to tritium.

No special equipment, procedures etc., beyond standard radioactivity
contamination control measures are used for handling of tritiated wastes,
and protective equipment for tritium is not normally required.

A variety of measures are employed to minimize tritium (and other
radionuclide) releases from the facilities and minimize occupational dose.

Releases from the WVRF are controlled by limiting incineration to waste
with a maximum dose rate of 60 mR/h. Tritium emissions are less than
0.1% of the Derived Emissions Limit (DEL) of 1.4×10^5 Ci/week.

Baler and compactor exhausts discharge directly into the building
ventilation system exhaust stack thus minimizing occupational dose.
Ventilation system and incinerator exhaust stack discharges are monitored
for tritium and gamma emitting radionuclides.

The Low Level Storage Building (LLSB) is also monitored for tritium and
gamma activity levels. The airborne tritium level in the LLSB is
typically 1 uCi/m^3. All waste is packaged in plastic or placed in
sealed containers to minimize releases. Releases of tritium from
rejected heavy water drums has resulted in airborne tritium levels of up
to 30uCi/m^3 inside the LLSB. Respiratory protection is used when
airborne tritium levels exceed 10 uCi/m^3.

Waste containing significant free water is not processed or stored at the waste site. Prior immobilization treatment would be carried out.

Waste site surface and subsurface water are monitored for tritium and gross beta gamma. Loss of storage structure integrity would be detected by such monitoring. Subsurface tritium concentrations range from 5 to 25% of the DEL of 5.5 mCi/m^3 while surface runoff tritium concentrations range from 1 to 20% of the DEL. Condensation and/or wash out of airborne tritium from the incinerator stack effluent, vapour from storage structures and the nine reactors located at the site contribute to the surface water tritium concentration. Tritium in subsurface water is generally attributed to contamination from surface water. Total drainage water volume is low resulting in low emissions of total activity and negligible hazard to the public. Ground water samples taken outside the site boundary indicate releases are less than 0.3% of the DEL.

2.0   Tritiated Waste Conditioning

A heavy water detritiation system is being commissioned at the Darlington Nuclear Power Station site and will be placed in service during 1987. The purpose of the plant is to reduce occupational dose from tritium and will extract approximately 20×10^6 Ci of tritium per year.

Table 2 shows the types and quantities of tritiated waste the operation and maintenance of the tritium removal facility is expected to generate (Reference 2).

TABLE 2

TRITIATED WASTES ASSOCIATED WITH TRITIUM REMOVAL FACILITY

Type of Waste	Activity Level	Estimated Max Volume of Waste
Liquid		
(a) Water	10^{-6} Ci/kg	200 m^3/a
(b) Electrolyzer Fluid (25% KOD)	0.5-1.0 Ci/kg	0.025 m^3/a
(c) Pump Oil	0-50 Ci/kg	0.075 m^3/a
Solid		
(a) Low Activity (Processible)	<0.2 Ci/m^3	132 m^3/a
(b) Medium Activity	0.2-100 Ci/m^3	0.4 m^3/a
(c) High Activity	>100 Ci/m^3	1.3 m^3/a

Some of the waste generated in the tritium removal facility will require conditioning and packaging to ensure storage criteria is met. Therefore, an extensive R&D program was initiated by Ontario Hydro to identify and develop the techniques and processes required to ensure waste isolation for 150 years (12 tritium half lives) (Reference 3).

Immobilization and encapsulation media have been investigated, specifically, organic compounds, inorganic hydrates and desiccants, cements, and their combinations. Waste conditioning requirements are based on the assumption that the waste conditioned for storage should not require further conditioning should it be moved to disposal. Two criteria were established for conditioning:

- daily tritium release from conditioned waste under leaching conditions not to exceed 10^{-3}% of initial activity,

- waste form integrity to be maintained for 150 years.

2.1 Liquid Waste

2.1.1 Aqueous

Cement based solidification agents, alone or with additives, were investigated to determine water retention on exposure to moist air at different temperatures. It was concluded that:

- Additives increased the rate of water loss and, in the case of epoxy resin, interfered with the cement hydration process

- Portland cement and high alumina cement are the most suitable absorbents for tritiated water, however, neither will retain tritiated water

Silica fume as an additive was found to:

- Effectively reduce the permeability of water through cement by hydrating to form silica gel in the pores of the cement.

- Be a desirable additive in the formulation to solidify electrolyzer fluid. It improves workability by reducing the setting time and reacts rapidly with the alkali to prevent the formation of products in the cement which can cause cracking and deterioration.

Solidified tritiated water waste forms produced with water extendible polyester (WEP), Portland cement and high alumina cement, with and without silica gel as an additive, exceeded the maximum leaching rate of 10^{-3}% per day. WEP and Portland cement/silica flume appeared to be the best solidification media.

The use of the above media and epoxy resin as containers for encapsulating the solidified waste form was investigated. It was determined that WEP was a significantly more effective encapsulant than Epoxy resin which was in turn more effective than Portland cement. However, this method of containerization of the solidified tritiated waste was quite difficult to implement and significantly increased the complexity of waste handling.

Although the leach rate from the waste forms encapsulated in WEP was acceptable, a simpler method of packaging the solidified waste in high density polyethylene containers was investigated and the following conclusions drawn:

- Tritium release from liquid aqueous waste containerized in polyethylene is lower than solidified waste using WEP or cement with a polyethylene container. However, solidification is still required to avoid dispersion should containment be breached.

- Losses from tritiated water solidified in WEP were twice as fast as from cement using identical containers. As cement was determined to be inferior to WEP without a container, it was concluded that organic components in WEP attacked the polyethylene, increasing its permeability to water.

- Tritium release behaviour is not quite linear with the inverse of container thickness because organics affect thin polyethylene containers to a greater degree than thicker ones.

- With respect to WEP solidified waste, a 1 cm thick polyethylene container is equivalent to a similar thickness container made from WEP.

2.1.2   Tritiated Oil

The tritium release rates of tritiated gas and water from oil were measured. The tritium gas was determined to be released at a rate of approximately 0.2% per day while there was no measurable release of tritiated water. Analysis of tritium species in the oil showed that the gas contaminated sample contained 42% volatile tritiated species and 58% non-volatile hydrocarbons. In oil contaminated with tritiated water, only 30% of the tritium activity is attributed to dissolved or suspended water, the remainder was incorporated into hydrophilic additives and corrosion inhibitors in the oil.

An oil sample contaminated with tritium gas and a sample contaminated with tritiated water were each containerized in 0.1 cm thick polyethylene containers and leach rates determined. Both samples exceeded the maximum leach rate of 10^{-3}% per day, the loss from the sample containing tritiated water being an order of magnitude higher than from the tritium gas contaminated sample.

Whereas the use of WEP with a high density polyethlyene container
resulted in an increase in the permeability of water by one order of
magnitude, the water permeability through high density polyethylene was
increased by 10 orders of magnitude in the presence of oil. Therefore,
it is necessary to keep oil away from the container walls in order to
realize the benefits of using high density polyethylene as a container
for tritiated waste oil. Methods for solidifying the oil in WEP or
cement or the absorption of oil onto vermicullite or Sorbent C, an
expanded perlite and fibrous wood material were studied to meet this
objective. Results indicated that both cement solidified oil or Sorbent
C absorbed in oil in a 1 cm high density polyethylene container would be
able to meet the 10^{-3}% daily tritium release target. Oil could not be
solidified in WEP except at very low waste loading.

2.2   Solid Waste

Non-processible medium and high tritium level waste will consist mainly
of discarded metal components. The distribution of tritium in the oxide
layer and bulk metal of copper and stainless steel tubing, removed from a
tritium handling loop, was investigated.

The copper samples appeared to have a fairly uniform specific tritium
activity, whereas in the stainless steel samples it was much much more
variable. Tritium in the surface oxide represented 29% of the total
tritium for copper and ranged from 52% to 80% for stainless steel.

The copper samples were leached for 150 days. Of the total tritium, 26%
was removed by the leachant in the first 7 days, thereafter there was no
measurable release. For stainless steel samples there was a rapid
release of tritium within the first 7 days which decreased to 0.1% per
day thereafter.

It may be concluded that surface tritium may be readily washed off and
that if tritium contaminated solids were allowed to stand, especially in
moist air, the tritium would be released to the environment through
isotopic exchange with atmospheric moisture.

Since the rate of diffusion of tritium from bulk copper is less than the
10^{-3}% per day no further conditioning would be required once the
surface tritium is removed. For stainless steel, the diffusion rate of
0.1% per day exceeds the 10^{-3}% maximum and therefore will require
conditioning.

Tritium release rates from polyethylene containerized tritiated stainless
steel samples, with and without WEP encapsulation, were compared.
Containment of samples in polyethylene was ineffective at retarding
tritium releases, the reduction factor being two. Encapsulating the
samples in WEP prior to containerization was very effective, the tritium
release rate being reduced to 10^{-4}% per day.

3.0 Fusion Reactor Wastes

The tritiated waste management activities and facilities described above will be suitable for much of the tritiated wastes arising from fusion test and power reactors. However, some components from the first wall, tritium fuelling and blanket systems will contain activation products and be highly contaminated with tritium.

Based on the results of the R&D to date, any delay in encapsulation and containerization of these wastes will result in the liberation of significant quantities of tritium gas and vapour. Tritiated wastes should be encapsulated and containerized for storage immediately. If not, they should be decontaminated and placed in an area equipped with a system to recover airborne tritium.

Fusion research and power facilities will require a designated waste processing and active maintenance facility and associated equipment. Such an area would be equipped to handle activated and contaminated components. Maintenance of tritiated components would be greatly simplified if they were decontaminated and/or degassed. Any facilities provided for maintenance could also be used during decommissioning.

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Purification System Filters → In-Station Shielding Flask → Drip-Can → Type 'B' Transportation Flasks → Truck → Storage

In-Station Filter Handling Room

Miscellaneous Incinerable Waste

Miscellaneous Compactable Waste — In-Station Transporter → Type A or LSA Packaging — Truck → Incinerator → Storage

Miscellaneous Non-Processable Waste

In-Station Solid Waste Management Area

Baler Compacter → Storage

Bruce NPD Waste Volume Reduction Facility

→ Storage

Spent Resin in D₂O Systems → I/S Flask → D₂O Recovery → I/S Flask → 'Type 'B' Transportation Flask → Storage

Pipeline

In-Station Wet Storage — Bulk Resin Container

Spent Resin in H₂O Systems — Pipeline

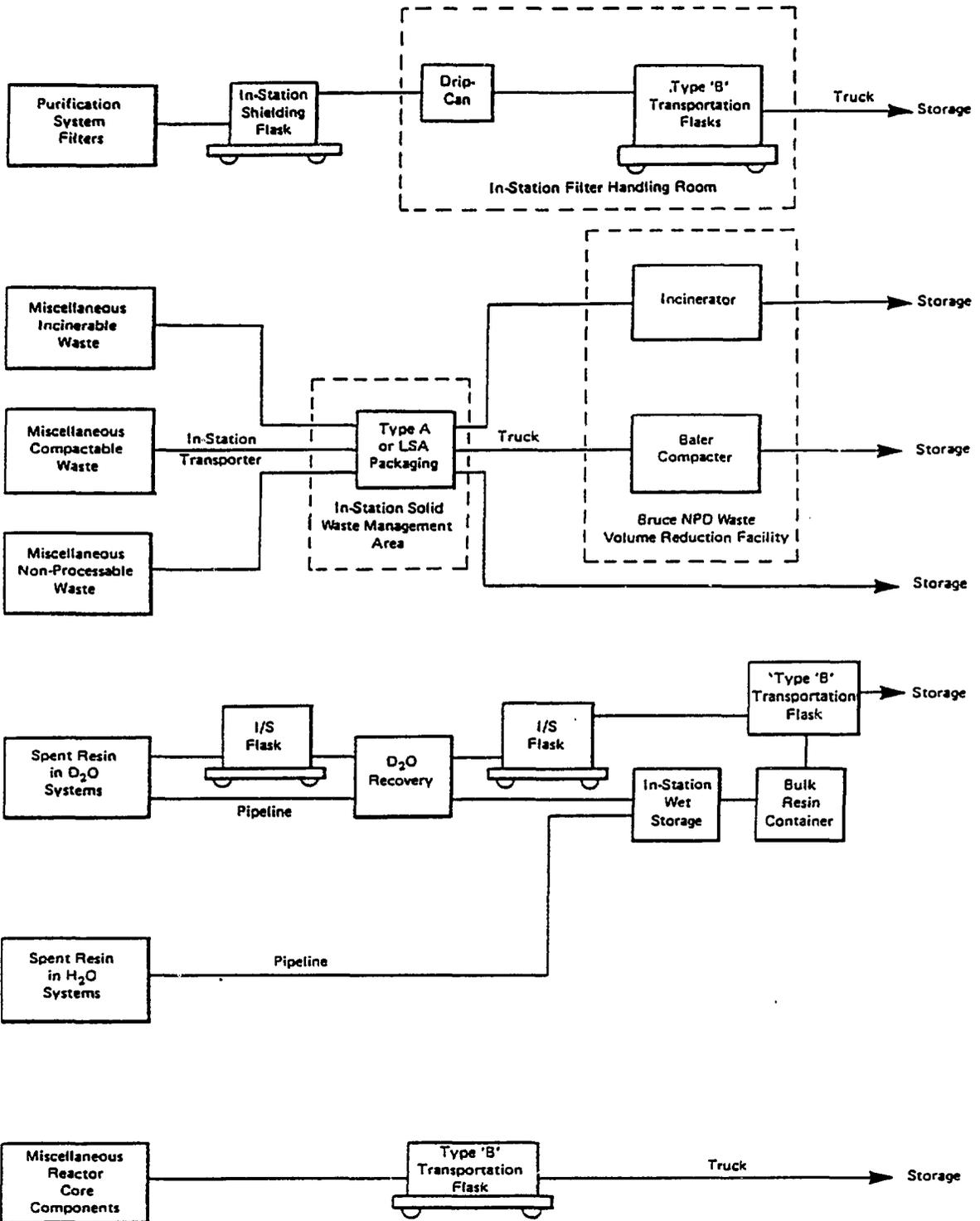Miscellaneous Reactor Core Components → Type 'B' Transportation Flask → Truck → Storage

FIGURE 1
Radioactive Solid Waste
Management Flowsheet