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HTR-N1 REVIEW OF EXISTING WASTE TREATMENTS

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HTR-N1 WP4 REVIEW OF TRADITIONAL TREATMENTS FOR OPERATIONAL WASTES

1.0 Introduction

Operational Wastes from typical UK nuclear power stations can be considered to arise mainly from three sets of activities. Each of these main activities in turn produces waste sub sets, broadly as indicated below

Ongoing power operations, giving rise to	- gaseous wastes - liquid wastes - spent ion resins - sludges - spent filters
Maintenance, giving rise to	- fuelling machine parts - control rod mechanisms - gas circulators, including contaminated oils and fluids - gas circuit parts
Refuelling Operations, giving rise to	- spent fuel - fuel element debris

The above will be generally applicable for all types of reactor, although for direct cycle machines, e.g. BWRs and designs proposed for HTRs, there will in addition presumably be wastes arising from the turbines and power circuits. From a UK perspective traditional treatments for these wastes have included the following as described under sections 2,3 and 4 below, dealing respectively with the Gaseous, Liquid and Solid wastes.

2.0 Ongoing Operations, Gaseous Wastes

In the UK, all operating power stations will have authorisation for gaseous discharges which will specify annual limits for tritium, S-35, I-131, Ar-41, C-14 and beta activity and weekly limits for C-14 and S-35. There will also be quarterly notifications for S-35, I-131, Ar-41, C-14 and beta activity. All discharges to atmosphere from approved outlets, which are defined by the ventilation system, are sampled during discharge and also an environmental monitoring programme will be carried out around the power station.

Potentially contaminated areas are ventilated by induced airflow drawn through filters. Separate systems exist for different parts of the station where necessary. The Contaminated Ventilation System will cover most of the Reactor Services Block. Rooms are usually vented through grills into ducting to relevant part of the system. High dust areas such as the irradiated Fuel Disposal Cell Debris Vault have additional particulate filters instead of grills. Sludge and sand/resin tanks

are vented by duct connection. In nearly all cases filtration before discharge will be through modular type HEPA filters. For areas where iodine activity is possible, iodine filters are also available.

A minimum of 99.99% of solid particulate is removed from the gaseous wastes through the HEPA filters. The discharges will be made to atmosphere at an elevated level, above the height of the reactor building, to ensure necessary dispersion.

Reactor coolant operations will also give rise to gaseous discharges, in this case predominantly CO_2. All controlled discharges of gas are made through the Reactor Blowdown System. This involves the use of primary stage filtration using the reactor by-pass filters, followed by HEPA filtration. Finally, gas is treated to remove any I-131 by the Iodine Absorption Plant. Prior to a planned and significant blowdown, a gas sample will be taken and analysed, and consultation with Health Physics may be a requirement.

Reactor coolant quality is continuously maintained including drying through gas dryers to remove moisture. During this process tritium and some S-35 is entrained in the liquid waste. Some S-35 will be removed by the charcoal in the Iodine Absorption Plant.

During reactor shutdown, maintenance and start up, extensive carbon dioxide purging is undertaken to remove as much air as possible, to achieve the required limit for the reactor coolant gas and limit the activity from activated argon and nitrogen.

The primary radionuclides in gaseous form will be tritium, C-14, S-35 and Ar-41. Metallic activation products are present in particulate form. Minor quantities of fission products such isotopes of krypton and xenon and volatile iodine are also present.

The operations of the auxiliary gas circuit and fuel route, parts of which are maintained in carbon dioxide atmosphere, will give rise to similar gaseous wastes as the reactor coolant system

In summary, the gaseous wastes are released to atmosphere after the due process of filtration through HEPA filters and adsorption plant. This method of treatment gives rise to consequent wastes, during operations and maintenance, the treatment for which is considered below.

It is difficult to envisage any different strategy for gas reactors, so that the principles for HTRs are likely to be similar to the UK gas reactors.

3.0 Ongoing operations – Liquid Wastes

Radioactive liquid effluent discharges in the UK are subject to authorisation limits similar to the gaseous wastes described above. Power stations are expected to use practical means to limit discharges and quarterly and annual notification of discharge levels of various nuclides are required. All discharges of radioactive liquids are made through approved outlets and are sampled prior to and during discharge. Also an environmental monitoring programme is carried out around the station.

Liquid wastes may be categorised in 3 or 4 types as follows.

- Active effluent arising from reactor, fuel route, and cooling pond operations, routine liquid waste operations, and also from equipment maintenance
- liquor arising from the reactor coolant system, particularly tritiated liquor from the gas drying system
- Radioactive waste oils from the operation and maintenance of gas circulators and from oil separators used in liquid waste processing

Each of these liquid waste streams are typically described and treated as follows

The active effluent treatment liquids will have relatively small amounts of tritium present, their activity usually resulting from the presence of activation products. The most radioactively significant activation induced nuclide will be Co-60, although discharges of this will be very low.

The liquid wastes of the active effluent treatment may themselves be categorized as, potentially oily wastes, non-oily wastes, soapy wastes, tritiated waste and chemical wastes from drains. Fission products are generally not present in UK gas reactor liquid effluents.

Potentially oily wastes will be passed through oil separators. A radioactive drainage system is used to collect and segregate liquid wastes. Filtration is used to remove particulate radioactivity as far as possible. The more active liquid wastes may be treated using non-regenerable ion-exchange units to reduce their dissolved activity. All liquid effluent is sampled and assessed before discharge through an approved route.

Once all treated liquid wastes from the active effluent system have been filtered and potentially oily wastes separated as above they are then transferred to the Final monitoring and Delay Tanks. When discharges are planned from the tanks, a sample is taken and analysis carried out prior to authorisation of the discharge.

The liquid wastes give rise to spent ion exchange resins, sand and sludge wastes, treatments for which are described under the solid wastes section below.

Active liquors will arise from the reactor coolant gas driers as they remove moisture from the gas circuit. The major radionuclide present in the gas driers regeneration effluent is tritium, although some S-35 is also present. Tritium is released into the coolant from a variety of sources mainly from fission and diffusion through the fuel cladding and from Lithium impurities in the graphite. S-35 arises primarily from activation of sulphur impurities in the graphite and reactor gas. Significant amounts of other radionuclides are prevented in this effluent, because it arises as condensate from the regenerated drier desiccant.

Tritiated water drained from the reactor coolant gas driers is accumulated in the Tritiated Water Storage Tanks. Sampling and analysis is carried out prior to discharge. The water is discharged along the same route as that used from the Final Monitoring and Decay Tanks and at the same time that these tanks are pumped out.

Radioactive waste oil will arise from the operation and maintenance of gas circulators and from the oil separators used in liquid waste processing. Other organic liquids may arise from decontamination processes and liquid radiochemical analysis.

The wastes arisng from the oil separators may contain the full range of activity found in the liquid waste treated in the active effluent treatment plant. Similarly, the oil from the gas circulators may contain any of the nuclides found in the reactor coolant. Additional organic liquids arise in the form of solvents from decontamination processes, or from liquid radiochemical analysis and may therefore contain a range of radionuclides.

All of the waste oils and organic liquids are expected to Low Level Waste.

The treatment of the waste oil is typically as follows

The oil is first collected in drums and transferred to a storage facility. The waste oil is then filtered and dewatered to minimise volumes and improve quality for subsequent incineration. The oil is then transferred to a storage tanks to take advantage of radioactive decay. Sampling and analysis is carried out to obtain an activity estimate. When suitable the oil is then further transferred to a (trace heated) tank and eventually burnt in a filtered incinerator.

It is expected that waste oils from HTRs could follow a similar waste treatment. However if magnetic bearings are used, as proposed in some HTR designs the quantity of oily wastes will be significantly reduced.

Other contaminated liquids are stored on site. They are incinerated on site, possibly with suitable solid materials such as sawdust.

4.0 Ongoing Operations – Solid Wastes

There are several solid active waste streams in UK gas reactors as follows

- Spent Ion exchange resins
- Sand
- Sludges
- HEPA filters
- Adsorption materials
- Fuel stinger (or element) debris
- Miscellaneous contaminated items
- Miscellaneous activated items
- Filter dust

- Desiccant
- Miscellaneous and trash LLW
- Decontamination arisings (this latter item also includes gaseous and liquid wastes)

The treatment of these various waste streams is as described below

4.1 Spent Ion Resins

As indicated above spent ion resins arise from pond water treatment and liquid effluent treatment. The spent ion resins are non regenerable proprietary polystyrene/divinyl benzene based material. At present only organic resins are in use, but it is possible that for future use inorganic resins may be used. The waste resins comprise both Low Level Waste and Intermediate Level Waste. The contamination present is mainly Fe-55, Cr-51, Co-60, Mn-54, S-35 and Ca-45. In the UK gas reactors the quantities of spent ion resins is not high, as might be the case in other reactor types, and therefore the incentives for volume reduction by incineration or pyrolysis is not great. Wastes will continue to accumulate in the present storage facilities. Wastes which are LLW may be retrieved during operations, encapsulated in cement in standard boxes and sent to Drigg, the UK's LLW waste repository. Waste which is ILW will be allowed to accumulate throughout operational life. After shutdown and defuelling these wastes will be retrieved, encapsulated in cement grout in NIREX approved containers, and placed in interim storage awaiting the availability of the UK repository.

4.2 Sand

The processing of active liquid effluent produces waste in the form of contaminated filter sand. Sand is used as the primary filtration medium in the active effluent treatment and pond water treatment. Backflushing of the sand pressure filters produces waste in the form of sand mixed with sludges. Major arisings of filter sand occur when filter clogging requires a change of medium. A significant proportion of the contamination present in sand is short lived radionuclides such as Fe-55, Cr-51, Co-60, S-35, and Ca-45. Wastes are considered to be mainly ILW. The treatment of the waste is as for the spent ion resins as above. Wastes will continue to accumulate in the present storage facilities. Wastes which are LLW may be retrieved during operations, encapsulated in cement in standard boxes and sent to Drigg, the UK's LLW waste repository. Waste which is ILW will be allowed to accumulate throughout operational life. After shutdown and defuelling these wastes will be retrieved, encapsulated in cement grout in NIREX approved containers, and placed in interim storage awaiting the availability of the UK repository.

4.3 Sludges

As described above, the processing of active effluent produces waste in the form of sludges. Sludges arise from the backwashing of filters in the active effluent treatment plant and pond water treatment plant. Oil contaminated sludges arise from the gas circulator operations and maintenance, and oil separators. The sludges comprise of solid particulate matter. A significant proportion of the contamination present is short lived radionuclides such as Fe-55, Cr-51, Co-60, S-35, and Ca-45. Wastes are considered to be both LLW and ILW. Wastes which are LLW may be retrieved during operations, encapsulated in cement in standard boxes and sent to Drigg, the

UK's LLW waste repository. Waste which is ILW will be allowed to accumulate throughout operational life. After shutdown and defuelling these wastes will be retrieved, encapsulated in cement in NIREX approved containers, and placed in interim storage awaiting the availability of the UK repository. In addition, some sludges are incinerated on site and the ash disposed to Drigg, the UK LLW repository, after high force compaction. These latter sludges are typically those arising from the reception tanks of the active effluent treatment plant.

4.4 Fuel stringer (or element) debris

In UK AGRs fuel stringer debris consists of miscellaneous activated components from refueling operations. In UK AGRs, typically eight fuel elements are held together in a vertical configuration by a tie bar and fixings. On defuelling the elements are separated at dedicated facilities at charge face. Except for the upper parts of the tie bar, which are not exhibited to high flux levels in the reactor and can be classified as LLW, all other fuel stringer debris is ILW. The waste is stored in dedicated dry vaults. The debris consists of a wide range of items such as tie bar croppings, tie bar end fittings and nuts, reflector assemblies, thermocouple wires and central inertia collectors and their contents. The accumulated wastes are made from stainless steel and graphite and cover a wide range of sizes. The central inertial collectors contain active dust composed mainly of carbonaceous and steel oxide based materials.

In the UK Magnox reactors, fuel element debris consists of magnox splitter debris, which are the cooling fins removed under water from the magnox cans to enable better packing for transport to Sellafield, and on some magnox reactors removed graphite fuel sleeves.

The treatment of AGR fuel element debris and magnox fuel element debris is different only in respect of the timing of retrieval. In Magnox stations the fuel element debris will be retrieved after shutdown and defuelling, but before preparations are put in hand for long term safestore. AGR fuel stringer debris is considered to be more stable than its magnox equivalent and therefore will be accumulated in the current storage facilities (which are usually part of the bioshield). During Stage 3 decommissioning (reactor dismantling) the wastes will be retrieved, sorted, optimally packed in NIREX approved containers and encapsulated in cement grout.

4.5 Miscellaneous Contaminated Items

Miscellaneous contaminated items arise during reactor operations and are accumulated on site. They occur as a result of station maintenance or repair activities, usually associated with the fuel route. Their activity levels are usually above LLW. A significant fraction of the activity is due to short lived radionuclides e.g. Co-60 and S-35. Waste which is LLW will be encapsulated in cement grout in approved containers and sent to Drigg for disposal as soon as is convenient. ILW will be stored for decay and decontamination where appropriate. The decontamination process could consist of high pressure water jetting, the wash water being sent to active drains system. Otherwise waste which is ILW will be stored in the vaults and retrieved in Stage 3 decommissioning (reactor dismantling), the wastes will then be sorted, optimally packed in NIREX approved containers and encapsulated in cement grout.

4.6 Miscellaneous Activated Components

Miscellaneous Activated Components (MAC) arise during routine reactor operations. MAC consists of neutron activated ex-reactor or ex fuelling machine waste. It will include such things as irradiated flux measuring instruments, flattening bars, parts of control rod mechanisms and control rods themselves. All wastes are considered as ILW and may be stored in a number of vaults. Their treatment will consist of retrieval in Stage 3 decommissioning (reactor dismantling) to take advantage of radioactive decay, sorting, optimally packing in NIREX approved containers and encapsulation in cement grout.

4.7 Filter Dust

These wastes arise from routine reactor operations. They accumulate in the catchpots of various filters. Filter dusts comprise the fine particulates which are backflushed off the gaseous system filters, particularly from the reactor blowdown system and the fuelling machine cooling systems. Their composition is a mixture of ex –stainless steel oxides, desiccant/catalyst fines, graphite dust and other carbonaceous material. The waste is mainly ILW. The proposed treatment is to leave them in place until Stage 3 decommissioning, to take advantage of radioactive decay, when the ILW will be encapsulated in cement grout in approved NIREX containers and dispatched either to the UK repository, or to temporary storage awaiting the completion of such a repository.

4.8 Desiccant

In UK gas reactors, desiccant waste arises from recharging the reactor coolant drying systems. The desiccant is discharged during reactor outages to the storage vaults. A number of different types of desiccant are in use. Typical desicant is Mobil Sorbead “R” silica gel beads. Radionuclide levels are low, except for tritium and S-35. The short half life of S-35 (87 days) means that its activity will be low after any appreciable storage period (currently many UK reactors operate for 3 years between outages). The level of tritium in the waste may well determine whether it is ILW or LLW. The proposed treatment is to leave the desiccant in situ in the vaults until Stage 3 decommissioning where stations have dedicated storage vaults and sufficient vault capacity for life time operation. The waste would than be treated if necessary to reduce activity levels and disposed to a suitable facility. Where either of these provisions is not available on site (i.e a dedicated vault or sufficient lifetime capacity), the treatment will be to decontaminate, probably encapsulate in cement grout, and dispose as LLW to Drigg.

4.9 High Efficiency Particulate Arrestor (Air), (HEPA) Filters

As described above, filtering of gaseous waste streams to remove particulate matter produces spent HEPA filter wastes. The HEPA filters are usually constructed from fibre glass filter material carried in a case. They will contain solid particulate comprising graphite dust and iron oxides. The waste is expected to be LLW in normal operation. There are two possible types of treatment. For lower activity filters, these will be incinerated, the ash drummed and compacted, and disposed as LLW to Drigg. Higher activity filters, and lower activity filters not suitable for

incineration, will be size reduced, drummed, high force compacted and disposed directly to Drigg as LLW.

4.10 Charcoal filters (Adsorption)

As described above, treatment of gaseous wastes to remove volatiles gives rise to spent charcoal filters (adsorption media). The Iodine Adsorption Plant is used to remove Iodine from reactor gas and also from fuel route ventilation sytems. The Iodine adsorption plant operates with a deep bed of activated carbon (charcoal) smeared with either potassium iodide or tetra ethylene diamine according to whether the plant is operating in carbon dioxide or air. Periodically the charcoal filter becomes exhausted and has to be replaced, giving rise to waste. The spent charcoal is expected to LLW in normal operation, the only significant activity being from radioactive iodine, S-35 and other volatiles. The proposed treatment is to remove the charcoal by vacuum into drums, store to allow for decay, then send to Drigg for disposal after high force compaction. A possibility is to volume reduce by incineration.

4.11 Spent fuel

In the UK gas reactors spent fuel is discharged from the reactor by the fuelling machine. In Magnox reactors the fuel passes to the fuel pond (after separation of graphite sleeves if necessary), the magnox splitters are removed as described earlier, and the fuel is held in the pond for 90 days. In the UK AGRs the fuel will pass to the cooling pond via the buffer store and irradiated fuel disposal facility. Following the requisite periods in the cooling pond, the fuel will be put into transport flasks and sent to Sellafield. At Sellafield the fuel is reprocessed. Approximately 3% of the spent fuel volume, when reprocessed, results in high level waste (HLW). The treatment for this HLW is currently to vitrify at Sellafield and place into a purpose built store. The intention is retain this HLW in the store for 50 years, to allow for cooling, then to place in the UK repository, when it is available.

For the UK's PWR, Sizewell B spent fuel is currently stored on site, under water. It is possible that a dry store will be built for the storage of the fuel.

4.12 Miscellaneous and Trash Wastes

These wastes include protective clothing and other items which may arise from activities within the controlled areas. They are all low activity. The wastes are identified in streams according to their radionuclide content. These low level wastes are classified as combustible or non combustible. It should be noted that not all combustible wastes can be incinerated, either because their activity leveles are too high with respect to discharge limits, or because their chemical content would cause a breach of environmental authorization limits. Non incinerable wastes can be either compactible or non compactible.

The methods of treatment for such wastes are as follows

The wastes are accumulated and segregated according to their chemical and physical properties.

LLW arisngs are processed, volume reduced if possible/appropriate and dispatched to Drigg as soon as possible.
Where possible, combustible wastes is incinerated, usually on site in the purpose built incinerators. Authorisation allowances limit the activity in the flue gas, and there will typically be monthly and annual limits in the wastes fed to the incinerators. The incinerators will typically include loading facilities, a combustion chamber, secondary after burning chambers, a de ashing screw, a heat exchanger, a wet scrubber and absolute filter leading to high velocity stack equipped for particulate and S-35 bubbler sampler. Te resultant ash is put into drums lined with a fireproofed polycotton ash bag for dispatch to Drigg for supercompaction and disposal.
Compactible waste is drummed and sent to BNFL for high force compaction at Sellafield.
Non compactibl waste is sent to Drigg for disposal
Waste not capable of being classified as any of the above is termed as non conforming wastes. This is decontaminated if possible, and stored on site.

4.13 Waste streams from decontamination

The wastes considered under this heading arise from the routine decontamination of plant items and equipment in Decontamination Centres and Active workshops. Decontamination is carried out either to allow equipment to be returned to service use or to reduce activity levels of waste to facilitate subsequent treatment and packaging. The most common plant items that are decontaminated are gas circulator components. Equipment items include protective air suits. Decontamination is only carried out where there is a clearly perceived net benefit in ALARP terms.

The wastes arising may be gaseous, liquid or solid, and is nearly always LLW. Gaseous wastes are generated as off gas in the decontamination process. Liquid wastes may be aqueous or in the form of contaminated organic liquids or oils. Solid wastes are either particulate matter or miscellaneous secondary wastes such as cleaning rags or brushes.

Gaseoues wastes are treated in the same manner as gaseous wastes from other streams, as described above. Aqueous wastes are segregated and flushed into the active drains system for treatment in the same manner as other active effluent as described above. Contaminated organic liquids are treated in the same manner as other organic liquids. Solid waste is treated in the same manner as miscellaneous and trash wastes.

5.0 UK waste management strategy

As can be understood from the above, the majority of UK waste management and disposal is based on encapsulation in cement grout, inside standard waste containers, prior to shallow land burial in the case of LLW, or likely deep disposal in the case of ILW. LLW waste disposal facilities exist at Drigg, which is operated by BNFL and is close to Sellafield. Facilities for final ILW disposal do not yet exist in the UK, but specifications for acceptance of waste in the eventual facility exist. NIREX is currently the governing body in the UK for disposal of ILW. When ILW waste is required to be treated and held in temporary storage, NIREX will assess the proposed form of treatment, provide guidance as necessary, and when NIREX is satisfied it will

issue a ”letter of comfort” to the owners of the waste. The letter of comfort will assure the waste owner that their waste will eventually be accepted by the ILW repository, when available. The proposed waste treatment methods are based on this strategy, the research and development carried out and described under section 6 below, and are governed by the decommissioning strategy put forward by the UK utilities as described under section 7.0 below.

6.0 Waste Treatment Methods

As will be appreciated from the above, the predominant treatment method in the UK is encapsulation in cement grout. Often the cement grout includes pulverized fly ash (PFA) from coal fired stations. As will be noted from descriptions given below, the cement grout has to be a carefully prepared and maintained mix of substances to ensure correct encapsulation. Prior to selection of cement as the encapsulation medium, several other options were researched in the UK. Given below are the options considered for encapsulation and the properties required of the eventual matrix. together with an outline of the R&D necessary.

6.1 Options and required properties for encapsulation

The forms of encapsulation researched in the UK are as follows

(i) cement

(ii) polymer

(iii) bitumen

(iv) resin sand

(v) vitrification

(vi) stone/ceramic

(vii) metallic combination.

For any encapsulation process, basic problems to be overcome are:

(i) floatation of smaller particles and dust, to the top of the grout

(ii) lack of penetration of the grout into the interspaces.

(iii) lack of homogeneity in the eventual matrix

A two stage encapsulation process may be used to overcome floatation problems. In the initial stage the grout or encapsulation medium covers the waste, in a drum or box, up to a few inches of the final surface. At this stage an anti flotation plate or a metallic grid may be used to retain particles. After the initial grout has set, a capping layer is then introduced. The two stage process ensures that particles are kept down. A separate lid can then be put on, or cast into the drum or box.

To ensure full penetration of the encapsulation medium the concentration of grout feed, or encapsulation mix has to be controlled and this is also necessary for ensuring homogeneity of the matrix. This is particularly the case for waste in smaller containers such as drums, where sacrificial paddles may be used to assist in the mixing process. The size of container may also affect mixing abilities. In the UK there may be a need to consider drums at 200 l and 500 l sizes, and 3 m^3 and 4 m boxes.

In the R&D programme it has been logical to carry out tests on unirradiated waste samples first, and indeed his has been done extensively in the UK. It is then important to understand the relevant differences between irradiated and unirradiated waste.

For encapsulation, research and development knowledge of the following factors are required:

(i) heat output from radioactive decay

(ii) radiation dose to the encapsulation matrix

(iii) heat released, and temperatures reached during the setting process.

In order to complete the assessment, it would also be necessary to know and model the conditions of storage in the repository, e.g. layout and density of containers, backfill medium used. Since the UK does not yet have an identified final repository for ILW, reasonable assumptions have to be made in these respects.

Encapsulation trials provide data on the following properties:

(i) mechanical properties – homogeneous matrix

(ii) mechanical stability

(iii) shrinkage/expansion testing

(ii) impact testing

(iii) resistance to radiation damage

(vi) leaching rates.

For the mechanical properties, it will be necessary to demonstrate that all the waste has been encapsulated. As a complementary issue, it will also be necessary to demonstrate that there is minimal voidage in the matrix.

The prime consideration for mechanical stability is that the matrix will have sufficient strength to withstand handling and transportation to the intermediate or final repository. The strength should also be sufficient to withstand accident conditions during transport, such as dropped loads. Moreover, this strength should ideally be achieved within a short time duration after setting. Once settled, the matrix should remain structurally and chemically stable throughout the life of storage and disposal.

Certain encapsulation processes, including cement, will be subject to dimensional changes resulting from hydration during curing. These changes should be monitored so that they may be compared with other mature structures of a similar kind, and which are known to be stable, and therefore give confidence that the waste matrix will remain stable.

With regard to thermal stability, it will first of all be necessary to consider whether there will be interim storage at or near to ground level. If this is the case, then according to location, the effects of winter freezing will have to be demonstrated. The effects of freeze and subsequent thaw will have to be determined, and the number of cycles for which this will happen. The freezing cycles should demonstrate that there are no adverse effects on the mechanical stability.

High temperature stability will also be required. The greatest temperatures conceivable will be from external fire hazard. The actual temperature reached will depend on the accident scenario assumed. In the UK, a pessimistic assumption would be 800°C for 30 minutes. This is the accident scenario to be assumed during transport of nuclear fuels.

An assessment of the specific heat and thermal conductivity of the matrix will enable heat build up from radioactive decay to be derived. For reasonable conditions of underground storage the heat build up is not expected to be significant.

Radioactive decay will take place in the body of the matrix, both in any interim and final storage. The cumulative effects of the irradiation on the stability of the structure will

require to be demonstrated. This has been assessed by subjecting samples to high levels of irradiation in order to simulate the long term effects of self irradiation.

Finally, and equally applicable to any of the disposal techniques, the concentration of waste material needs to be calculated to ensure that surface dose rates from the container satisfy regulatory needs in respect of transport, intermediate and long term storage.

6.2 Cement encapsulation

Of all of the encapsulation matrices researched, in both the UK and some other countries, cement has emerged as the preferred option. It is proven technology, fire resistant and does not produce high temperatures in the curing/setting period and is economic. It provides a high strength matrix (after curing), radiation stability, self shielding, and the high pH value will be an advantage in any final deep repository.

Amongst the first operational wastes in the UK from shutdown power stations, will be the wastes retrieved from the vaults at Berkeley nuclear power station. These wastes are being encapsulated by cement in standard NIREX containers, and will be held on site as ILW in a temporary (i.e up to 50 years) store pending the availability of a UK permanent repository. Irradiated wastes have also been encapsulated arising from the dismantling of the Windscale AGR (WAGR). WAGR is the pilot decommissioning project in the UK for the fleet off gas cooled reactors. The waste treatment and packaging route at this site has long been established and in use. WAGR wastes are being packaged in shielded concrete boxes, with Ordinary Portland Cement (OPC) and Pulverised Fly Ash (PFA) grout/matrix. The PFA in the grout and matrix offers very good protection against subsequent fire hazards. In the case of LLW the boxes are transported to the UK's Low level disposal facility at Drigg. ILW is being held in intermediate storage (up to 50 years) local to WAGR in a custom built store. Although the components being treated today at WAGR are decommissioning wastes, in the early days of the programme operational wastes were disposed of in the same manner.

6.3 Polymer encapsulation

Spent ion resins at the UK Trawsfynydd power station have been encapsulated in polymer in a custom built plant at the site. The chemical materials for the polymer matrix are supplied by the Dow Company. Although the process was originally intended for use with sea dumping, the plant was recommissioned, and polymer encapsulation has been accepted by NIREX for sub-surface disposal. The major advantages of polymer encapsulation are that it is a simple process, to proven technology, in a low temperature process. It provides a good strength matrix and offers low permeability. The disadvantages are (compared to cement) that is relatively expensive, is a possible fire hazard and has reduced radiation resistance. It should be noted that Trawsfynydd is

unique amongst UK nuclear stations. It is located inland, rather than at a coastal area, and is cooled by water from Lake Trawsfynydd rather than sea water. Allowable discharges to the lake are therefore considerably less allowed to open sea. For this reason the quantity of spent ion resins is somewhat greater than normal and this in turn led to different treatment methods. No other UK station uses polymer encapsulation.

6.4 Bitumen encapsulation

Bitumen encapsulation has been researched in the UK and elsewhere. It has good leaching resistance and can accommodate any initial water present. Another advantage of bitumen is the high ratio of waste to matrix (up to 4 parts waste to 1 part bitumen), thus ensuring relative waste volume minimisation. Although the process was researched extensively in UK and France, it has never been fully utilised in either country, and is now abandoned in both countries.

The bitumen process has been used in Japan for waste treatment, although not necessarily for power stations as yet. It is used eastern European countries for treatment of spent ion resins and in Canada also.. The obvious disadvantage with bitumen is the potential for fire hazards particularly where the waste itself is an oxygen or potential oxygen donor. In these latter circumstances, a fire hazard is possible even after the setting process is complete. Indeed at the PNC (now JNC) waste treatment works at Tokai, Japan, a bitumen explosion/fire did occur in 1997, with some release of radioactivity. There was also a bitumen fire at the Karlsrhue research facility in Germany.

6.5 Resin sand

Resin sand matrices have been considered by a number of countries. It is no longer considered as a preferred option. There are doubts that the sand grains can penetrate through the matrix and porosity in the raw waste form. Also the encapsulation process is complicated and involves high temperatures.

6.6 Vitrification

Vitrification, or glass encapsulation is being used for high level wastes in the UK and other countries. However, it is not thought suitable for the volume requirements of wastes for low and intermediate level. The process requires fine control and involves high temperature processing, which, if the process is not sufficiently controlled, can lead to cracks in the matrix structure due to differential thermal expansion. In turn the cracks can offer leach paths from the waste. The process has however matured in recent years for HLW, although problems still exist with the process.

6.7 Other encapsulation media

Other encapsulation media that have been considered include:

(i) stone/ceramic

(ii) metallic combination.

Stone/ceramic require high temperatures and pressures. Metal alloys with low melting points are expensive. It is considered that these last two options would require extensive development.

Finally, it should be noted that combinations of the above encapsulation techniques have been proposed. Lithuania, for example, has considered the study of ceramic-glass encapsulation for graphite, covered by an outer coating of polymer for shock absorbtion.

6.8 Sea dumping

All countries belonging to the European Community have accepted a moratorium on sea dumping, effectively since 1984. The UK accepts that this moratorium could be indefinite, but reserves the right to continue research and development and to re-open debate if the political climate were to change. In reality, it is more likely that a worldwide ban will be accepted by all countries.

At the time of the moratorium, both the governments of the UK and France believed that sea disposal was probably the best practical environmental option (BPEO) for disposal of all but high level wastes. Both polymer and cement encapsulation had been accepted for sea disposal. Where such waste is disposed in deep water the dilution factor would be very large, and even this would not take place for many years, after breaching of the containment and slow leaching outwards. The resulting additional radioactivity is likely to be undetectable against background levels.

6.9 Volume reduction techniques

In the treatment of operational wastes in the UK it can be seen that volume reduction techniques have so far extended to

i) incineration in the case of some oily wastes, sludges, low activity filters and trash wastes. Note that incineration is subject to environmental controls which are becoming ever more stringent, and which ultimately may cause a review of this method of volume reduction.

ii) Low force compaction, typically available on site for certain drummed wastes

iii) High force compaction, available at the LLW disposal facility at Drigg, and planned for use with incinerated sludge and filter ash, high activity filters not incinerated, and charcoal filters.

Note that, under the UK decommissioning strategy, as described below, a substantial part of the operational wastes are not dealt with until reactor dismantling, many years after reactor shut down. The benefit and incentive for considering volume reduction techniques for such components is therefore minimal at this stage.

7.0 The UK decommissioning strategy

The UK waste management strategy, is in part derived from the UK decommissioning strategy. The salient aspects of the decommissioning strategy are,

(i) After shut down remaining spent fuel is despatched from site as soon as possible

(ii) Operational wastes which are considered to potentially mobile are then immobilised as soon as all fuel has been despatched from site. The definition of mobile wastes not exact, but would include all liquid wastes, wastes which may come into contact with water.

(iii) When all the above activities have been complete the UK gas reactors are planned to have an 85 year (70 year for economic assessment) safestore period. Wastes which in vaults within the safestore enclosure, and are not considered to be potentially mobile, e.g. miscellaneous activated components, are treated in the same way as decommissioning wastes, during reactor dismantling.

Thus it can be appreciated that the UK decommissioning strategy has in part influenced the UK waste management strategies, which in turn have influenced the treatment of the various waste streams.

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Summary

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This report serves as information for the members of WP4 of the HTR-N (EC-Project: FIKI-CT-2000-0020) and is intended as an input for the SINTER Network.
It is the contribution of Framatome ANP GmbH to Deliverable 4.1 of WP4 of the HTR-N.
Remark:
Some of the given results depend strongly on the design of the systems, e.g. the release of gaseous materials is calculated without any filtering during normal operation.
In the case of an HTR with a direct cycle (PBMR or GT-MHR) you have to consider reflector graphite as solid waste and waste coming from the „turbine side“.

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List of revisions

Rev.	Date	Scope of revision	Section/Page

Introduction

The HTR-Module power plant (Siemens design of the 80ies) is a thermal power plant for the cogeneration of electricity and process steam. The process steam can be used for a wide variety of applications in the chemical industry, for example, or for tertiary oil extraction.
The place of the fossil-fired heat source of conventional power plants is taken in the HTR Module by two nuclear steam supply systems (modular units). Each modular unit comprises one high-temperature reactor, one steam generator and one primary gas blower.
The heat liberated by nuclear fission in the HTR is transported from the reactor through a coaxial duct to the steam geneator by the primary coolant, helium, which is circulated by the primary gas blower. In the steam generator, the heat is transferred to the water / steam conversion system which is designed and operated as a purely non-nuclear plant.
Auxiliary and supporting systems connected to the primary system are provided for operation of the reactor: furthermore, safety of the reactor is assured by systems which fulfill the task of keeping loadings on components and structures within acceptable limits under accident conditions and which minimize the impact of accidents on the operating personnel and the environment. The reactor auxiliary systems are installed in the reactor building and in the reactor auxiliary building.
The cooling loads in the power plant are served by closed cooling water systems which remove the rejected heat to service water systems.
Radioactive Materials in the Primary System and in Connected Systems

Of all the radioactive materials produced in the HTR Module power plant, the overwhelming majority is retained in the fuel. The U02 fuel in the form of approx. 0.5 mm diameter spherical particles is enclosed in an extremely effective barrier, the so-called TRISO coating of pyrocarbon and silicon carbide embedded in the inner zone of the graphitic fuel sphere which is surrounded by an unfueled shell.
If radioactive materials escape from the fuel elements or are produced in their proximity, they are for the most part retained in the primary system which, together with the directly connected auxiliary systems, forms a closed system.
The release of radioactive materials from the HTR Module power plant is only to be expected, if they leave this system, e.g. on regeneration of the helium purification system or as the result of primary coolant leaks.

Primary System

Fission Products

Causes for very low fission product release from the fuel elements during power plant operation are

- The low fuel fraction which is not enclosed in intact silicon carbide layers because of manufacturing defects
- The low irradiation-induced failed particle fraction and
- Negligible fission product diffusion through the intact coatings of the coated particles.

Considering all noble gas and iodine isotopes with half-lives of more than 15 min, the following release rates per unit of thermal power are obtained: noble gas release rate 9.1 x 10^8 Bq/h MW and iodine release rate 2.5 x 10^8 Bq/h MW. Radionuclides with shorter half-lives are irrelevant for the release of radioactivity to the environment since they decay in the plant.

Radioactive Materials in the Primary System of one Modular Unit

	Release Rate $\frac{\text{Bq}}{\text{h MW}_{th}}$	Steady-State-Coolant Activity in Bq[1)]	Surface Activity after 32 EFPY in Bq

Total Noble Gases	1.6 E+10	8.8 E+11	
Total Iodine	2.5 E+8	2.2 E+9	5.1 E+11
Total Short-lived Solids		1.8 E+10	2.1 E+11
Total Long-Lived Solids	8.9 E+4	8.2 E+5	2.6 E+12
H3 C14	2.8 E+7 3.0 E+5[2)]	1.1 E+11 1.2 E+9	

1) Relative to 15.000 m^3 of helium (20°C, 1 bar)
2) Primarily activation of N14

Tritium

Because of its ability as a hydrogen isotope to penetrate metallic walls at high temperatures, tritium (H^3) is primarily of interest for contamination of the water/steam cycle. It is produced both by fission in the fuel and by neutron capture in lithium impurities in the fuel element and reflector graphite and in the H^3 fraction of the primary coolant. By contrast, the production rates in the boron of the absorber system and the outer boron-containing reflector zones are negligible.

On account of burnup effects the power-dependent integral H^3 production rate drops from an initial 1.4×10^8 Bq/h MW to a quasi-equilibrium 5.5×10^7 Bq/h MW within a few years. Of the tritium produced in the fuel elements and reflectors, only a fraction is released into the primary coolant as a result of diffusion and storage mechanisms. The power-dependent release rate is therefore initially 2.8×10^7 Bq/h MW and at equilibrium 1.8×10^7 Bq/h MW.

Carbon 14

In high-temperature reactors, carbon 14 (C14) is mainly produced by neutron capture in nitrogen impurities in the fuel elements and primary coolant, as demonstrated by AVR operating experience. C13, large quantities of which are present in the graphite structures of the fuel elements and reflectors, only makes a comparatively low contribution to the integral C14 production rate. Transfer of AVR experience to the power-dependent C14 production rate in the primary coolant of the HTR Module power plant results in a value of 3×10^5 Bq/h MW.

Corrosion products

Since the inert gas helium is used as the primary coolant, corrosion and hence activation of metallic primary system surfaces only plays a minor role as source of radioactive materials in the primary system.

Distribution of the released radioactive materials in the primary system

The radioactive materials released into or produced in the primary coolant are transported from the core region to the primary system by the primary coolant flow. The activity decreases because of radioactive decay, separation out of the bypass flow of primary coolant through the helium purification system and deposition an surfaces to which the primary coolant has access.

The bypass flow of one module unit to the helium purification system amounts to 5 %/h. Since the activity retention efficiency of the helium purification system is practically 100 %, a purification constant of 0.05/h is obtained for the primary coolant. For solid fission products, metallic surfaces, particularly those of the steam generator and the colder graphite structures in the reactor, form the major traps for deposition. Operating experience, e.g. with the AVR, and calculations show that almost total accumulation on primary system surfaces is to be assumed in order to establish the deposited activity. Depending on the radionuclide, the expected deposition rates are between about 50 % and 90 % per primary coolant cycle. In order to establish the design activity of the primary coolant with regard to solid fission products and iodine isotopes, a deposition rate of 10 % per primary coolant cycle was conservatively assumed.

Helium Purification System

The helium purification system consists of three trains. One purification train is allocated to each of the two modules. The additional train is available when one of the module-dedicated trains is out of operation for reasons such as regeneration operation or repairs.
Contaminants are removed in stages from the purification flow extracted from the primary system. The main stages in the separation of radioactive materials are:

- Dust removal filters for separation of radioactive materials not in the form of noble gases (aerosols)
- Molecular sieve for adsorption of tritium, chemically bound in HTO, and C14, chemically bound in CO_2. The noble fission gases also experience a certain delay here, resulting in the accumulation of solid noble gas decay products
- Cryogenic adsorber for adsorption of noble fission gases and for adsorption of tritium and C14, chemically bound in CH_4. The solid radioactive materials resulting from decay of the adsorbed noble fission gases are also retained.

The following table lists the saturation activities of one train of the helium purification system as well as the most important purification stages.

Radioactive Materials in an Helium Purification Train

(Design Data)

	Saturation Activity in Bq of:			
	Dust Removal Filter	Molecular Sieve	Cryogenic Absorber	Total Acitvity in Bq of one Purification Train
Total Noble Gases		1.2 E+10	2.3 E+12	2.3 E+12
Total Iodine	1.1 E+9			1.1 E+9
Total Noble Gases Decay Products	4.7 E+8	1.7 E+9	2.7 E+10	2.9 E+10
Total long-lived Solids	5.8 E+9[1)]			5.8 E+9[1)]
H3 C14		4.6 E+12 5.9 E+10[2)]	9.3 E+11	5.6 E+12 5.9 E+10

1) After 32 equivalent full-power years
2) Not calculated, 100% of the C14 activity in the modular sieve

Regeneration systems

The molecular sieves and cryogenic adsorbers of the helium purification system are connected to separate regeneration systems. The tritium (HTO) obtained during regeneration of the molecular sieves is condensed out of the regeneration gas and temporarily stored pending disposal inside the plant in the tank for condensate collected in operation. The tritium present in CH_3T in the regenerate from the cryogenic adsorbers is converted to HTO by catalytic combustion of the methane, condensed out and similarly transferred to the tank for condensate collected in operation.
Gaseous radioactive materials such as the noble fission gases, the C14 chemically bound in CO_2 and the unseparated tritium are conveyed to the storage tanks for radioactively contaminated helium.
The radioactively contaminated helium resulting from regeneration of one purification train is collected in the storage tanks for radioactively contaminated helium after approx. 12 h at the earliest. The nuclide-specific activity which is then present in these tanks is apparent following calculation:

Accumulated Activity in the Storage Tanks for Radioactivity Contaminated Helium after one Regeneration Cycle (integral decay period 12 h)

Nuclide	Accumulated Activity in Bq after one Regeneration Cycle
Kr 83m	2.7 E+7 [1)]
Kr85m	2.8 E+9
Kr85	1.6 E+10
Kr87	9.7 E+6
Kr88	1.4 E+9
Xe131m	2.1 E+10
Xe133	2.0 E+12
Xe135	3.8 E+10
Total Noble Gases	2.1 E+12
H3	1.1 E+11
C14	5.9 E+10

1) 2.7 E+7 is equivalent to 2.7 x 10^{+7}

Gas evacuation system of the helium purification system

In the course of the regeneration process, the gas evacuation system of the helium purification system evacuates the components to be regenerated and the regeneration loop itself. The evacuated gases are pumped into the storage tanks for radioactively contaminated helium. Their activity burden is contained in the numerically listed data, above.

Gas evacuation system for fuel handling equipment

The fuel element charge locks are evacuated to prevent air inleakage into the primary system. The evacuated air, which can contain traces of radioactive materials, is purified in the system's filters and mixed with the exhaust air flow from the controlled area.

Gas evacuation system of the primary system

The purpose of the gas evacuation system of the primary system is to evacuate the latter prior to initial filling. It is used to evacuate the upper section of the reactor pressure vessel before and after maintenance and repairs. The primary coolant extracted prior to maintenance work is conveyed to the dump system for the helium supporting systems and fuel handling equipment. The evacuated air obtained after maintenance work is monitored for activity and, if necessary, is transferred to the stack via the exhaust air filter unit which is connected during maintenance and repair work. The expected quantities of activity are low.

Dump system for helium supporting systems and fuel handling equipment

The helium pressure in the following systems to be reduced for operational reasons from operating pressure to approx. 1 to 2 bar:

- Helium purification system before regeneration
- Sampling system (gas analysis system)
- Fuel element charge and discharge locks .

Depending an the activity or chemical contamination, the dump gas is transferred to a train of the helium purification system or to the storage tanks for radioactively contaminated helium. Tables above give the activity originating from the dump system.

Air Activation

Escaping neutrons can activate air in the primary cavities in the vicinity of the reactor pressure vessel. The principal product of air activation is Ar41 which has a half-life of 1.83 h and is produced at a rate of 8 x 10^{9} Bq/h per modular unit. Since the HVAC systems keep the primary cavities at a subatmospheric pressure lower than that in the surrounding equipment and service compartments, contamination of air with Ar41 is restricted to the primary cavities. There is no access to these cavities during reactor operation.

The air volume of one primary cavity is approx. 2000 m^3. In order to maintain subatmospheric pressure, about 2000 m^3 are extracted each day and transferred to the vent stack. The equilibrium activity concentration of Ar41 amounting to 9.5 x 10^6 Bq/m^3 therefore develops in the primary cavities. After reactor shutdown the activity concentration drops rapidly because of the short half-life and is therefore of no significance for accessibility of the primary cavity.

Radioactive Materials in Cooling Water Systems

Operational Component and Secured Cooling System

The closed cooling water systems are connected to cooling loads at which radioactive materials are produced by neutron capture and at which there is always the possibility of an inleakage of radioactivity from primary coolant-conveying systems. This primarily applies to the following cooling loads:

- Cavity coolers
- Support coolers of the reactor pressure vessel
- Primary gas blower coolers
- Heat exchangers of the helium purification system, the helium supporting systems and the fuel handling equipment

Radioactivity in the loops of the closed cooling water systems is governed in the cavity cooler area by activation of structural materials followed by corrosion and by activation of the water and any impurities in it. Model calculations provide the following reference data for radioactivity conditions in the closed cooling water systems:

Of the corrosion products, the cobalt isotope Co60, at approx. 3 x 10^8 Bq, constitutes the highest accumulated activity in the closed cooling water systems after 32 years of power plant operation. The total saturation activity of the other corrosion products such as Fe55, Fe59, Mn54, Mn56 and Ta182 is lower by about one order of magnitude. These numerical data are based on a corrosion rate of 10^{-3} mm/a in the cavity coolers.

The contribution of the nitrogen isotope N16 to radioactive releases is insignificant because of its short half-life of 7.13 s.

Since neither conditioning nor blowdown of the loops of the closed cooling water systems is provided, radioactive materials can only be released from these systems through breaks.

The activity concentration in the loops of the closed cooling water systems is monitored.

Service Water Systems for Operational Component and Secured Cooling System

The service water systems are separated from the loops of the closed cooling water systems by the closed cooling water system heat exchangers. Therefore they do not normally contain radioactive materials.

Water/Steam Cycle

The content of radioactive materials in the water/steam cycle is extremely low. It is governed by

- Tritium permeation from the primary system, and
- Activation products forming in the steam generator region.

Break-induced inleakage of radioactive materials from the primary system into the water/steam cycle is impossible during normal operation and anticipated operational occurrences because of the lower pressure in the primary system. Leakages through steam generator tubes are concomitant with a moisture build-up in the primary coolant. This is detected, causing shutdown of the affected module and uncoupling of the defective steam generator from the water/steam cycle.

Because of its property of being able to pass through metals at high temperatures (permeation), tritium enters the water/steam cycle even through intact steam generator tubes, governing the activity inventory in the cycle. Calculations indicate that 8.4 x 10^7 Bq enter the water/steam cycle each hour through the two steam generators of the HTR Module power plant. The maximum activity concentration is reached during operation

with the water/steam cycle closed.
The activity concentration in the water/steam cycle adjusts to approx. 6.3 x 10^7 Bq/Mg.
The thermal neutron flux which reaches the steam generator via the gas duct pressure vessel between the reactor and steam generator pressure vessels amounts to approx. 10^7 $s^{-1}cm^{-2}$ at the steam generator tube bundle inlet. The neutron flux decreases rapidly in the tube bundle and the average is therefore lower by about one order of magnitude. The saturation activity resulting from neutron capture reactions integrated over the tube bundle is estimated as being less than
2 x 10^{10} Bq on the basis of the main activation reactions. Corrosion-induced metal thinning causes less than 1 x 10^7 Bq per year to enter the turbine building with the main steam from the two steam generators, primarily during startup and shutdown operation.
Trace impurities in the feedwater and conditioning additives also undergo activation in the steam generator to a certain extent. Extrapolation of the integral activity concentration of all radioactive materials in the main steam, as measured in the AVR during power plant operation, to the HTR Module power plant yields a value not exceeding 0.4 Bq/Mg.
Radioactive Waste and Radiological Impact on the Environment

General

The overwhelming majority of the radioactive materials produced in the HTR Module power plant is retained in the coated particles of the fuel. If radioactive materials escape from or are produced in the proximity of the fuel, they are for the most part retained in the primary system and systems directly conneeted to it. Only a very small fraction of the radioactive materials is released with the exhaust air or liquid effluents to the vicinity of the power plant.
Gaseous radioactive waste generated in power plant operation is collected in tanks. After a delay, it is released from the approx. 60 m high vent stack together with the radioactive materials entrained as a result of leakage into or activation in the exhaust air from the reactor building and the reactor auxiliary building. Releases are monitored.
All radioactively contaminated water collected inside the controlled area is transferred to the liquid waste system where it is treated in such a way that it can be released to the environment in compliance with statutory regulations. The aqueous residues left in the liquid waste system are processed in accordance with provisions for the storage of low and intermediate-level waste and transferred to drums.
Condensate containing tritium from the helium purification systems and helium supporting systems is temporarily stored in the reactor building in tanks for condensate collected in operation.
Solid radioactive waste generated in power plant operation and discarded contaminated parts are also conditioned and temporarily stored in the reactor auxiliary building.

Liquid Waste

(Tabs. 4.2.1/1 to 2)

1.1.1 Radioactive Releases

Liquid waste generated in the controlled area of the HTR Module power plant is collected and processed. This liquid waste includes washroom and shower water, liquid waste from laboratories (samples), water collected after any decontamination operations and sump water from the building drain system.
Table 4.2.1/1 lists the expected quantities and corresponding activity concentrations.
Liquid waste is collected in four tanks, each having a capacity of 20 m^3. Postulating an activity concentration of 3.7 x 10^7 Bq/m^3 (10^{-3} Ci/m^3) the activity inventory (without tritium) in the tanks is obtained as approx. 3.0 x 10^9 Bq (0.08 Ci).
An evaporator unit and a centrifuge unit are available for liquid waste processing. After processing the liquid waste is transferred to three monitoring tanks, each having a storage capacity of 20 m^3. Pursuant to KTA 1504, the water content of a monitoring tank may be released if its activity concentration is less than 1.85 x 10^7 Bq/m^3 (50 x 10^{-4} Ci/m^3). The water must be processed again if the concentration is higher. The average concentration of the liquid waste will be much lower than this release limit. Measurements carried out for the

purposes of decision-making are conducted in accordance with KTA 1504 before any release of liquid effluents.
In order to balance the annual radioactive releases, a sample is taken from each batch to be released and is subjected to nuclide-specific evaluation. In addition, a representative sample from each batch is retained as evidence.
Furthermore, as an additional safety measure, the liquid effluents being pumped out of the monitoring tanks are monitored. An activity measuring point (integral gamma measurement) in the discharge line immediately stops discharge if the radioactive release is in excess of the given limit of 1.85×10^{7} Bq/m^3 (5×10^{-4} Ci/m^3) or if the measuring instrument fails.
The liquid effluents are pumped out of the monitoring tanks and mix with the discharge receiving water.
Calculations for quantitative analysis of the radioactive releases are based an a release concentration (without tritium) after purification of less than 3.7×10^{5} Bq/m^3 (1×10^{-5} Ci/m^3). An annual radioactive release of approx. 1.85×10^{9} Bq/a (0.05'Ci/a) (nuclide mixture without tritium) is expected for an estimated liquid effluent volume of approx. 14 m^3/d.
On account of leakages in the turbine building, the predicted annual tritium release rate in turbine building effluents is approx. 6×10^{11} Bq (16 Ci) for operation with the water/steam cycle closed. The calculation is based an a 4:1 distribution of the tritium release over the liquid effluents and exhaust air pathways.
In order to envelop not only average operating conditions but also unusual occurrences which may occur in individual operating years (e.g. large-scale repairs, increased extent of inservice inspections, frequent load changes), a release rate of

- 9.25×10^{9} Bq/a (0.25 Ci/a) for the nuclide mixture without tritium, and
- 3.7×10^{12} Bq/a (100 Ci/a) for tritium

is included in the licence application.

Table 4.2.1/1
Origin of Liquid Waste

Origin of liquid waste	Quantity produced Standard/max.	Activity Bq/m^3 (Ci/m^3)
Decontamination and laboratory water	1 to 3 m^3/d 480 m^3/a	Up to 3.7 x 10^7 (1 x 10^{-3})
Sump and leak-off water	1 to 10 m^3/d 900 m^3/a	Up to 3.7 x 10^6 (1 x 10^{-4})
Laundry water	3 to 9 m^3/d 1440 m^3/a	3.7 x 10^4 (1 x 10^{-6}) to 3.7 x 10^7 (1 x 10^{-3})
Shower and washroom water	5 to 12 m^3/d 2220 m^3/a	Normally inactive 3.7 x 10^4 (1 x 10^{-6}) to 3. 7 x 10^6 (1 x 10^{-4})
Total effluents	5000 m^3/a	Average 3.7 x 10^6 (1 x 10^{-4})

The annual average is based an the assumption that the standard quantity is produced on 300 days and maximum quantity an 60 days.

Table 4.2.1/2

Radioactive Releases in Liquid Effluents and Activity Concentrations at the Point of Release (Point of Use)

Nuclide	**Fraction of nuclide mixture acc. to** [1)]	**Release based on a nuclide mixture acc. to** [1)]		**Activity concentrations at the point of use for mixing with 30 m^3/s of the average run-off of the discharge receiving water**	
	%	Bq/a	(Ci/a)	Bq/m^3	(Ci/m^3)
Co 60	24.0	2.2 E+9[2)]	(0.060)	2.41	(6.51 E-11)
Sr 90	0.5	4.8 E+7	(0.0013)	0.05	(1.33 E-12)
I 131	5.0	4.8 E+8	(0.013)	0.49	(1.33 E-11)
Cs 134	15.0	1.4 E+9	(0.038)	1.48	(4.00 E-11)
Cs 137	55.0	5.1 E+9	(0.138)	5.44	(1.47 E-10)
Ag 110m	0.5	4.8 E+7	(0.0013)	0.05	(1.33 E-12)
Total					
Mixture	100	9.3 E+9	(0.250)	9.92	(2.68 E-10)
H 3	100	3.7 E+12	(100)	3960	(1.07 E-7)

1) Model calculation

2) 2.2 E+9 is equivalent to 2.2 x 10^9

Exhaust Air

1.1.2 HVAC Systems in the Controlled Area

The gaseous and aerosol radioactive materials released from the vent stack during normal operation and anticipated operational occurrences primarily originate from the following sources:
a) Activation of the air in the reactor cavities
b) Leakages from the primary system and connected primary coolant-conveying systems
c) Releases from the storage tanks for radioactively contaminated helium
d) Releases from the gas evacuation system for fuel handling equipment
e) Emissions during maintenance and repairs to opened contaminated systems

Re a)

Activation of the air as the result of neutron capture only occurs inside the primary cavities in the vicinity of the reactor pressure vessels. Mainly Ar41 is formed (see Section 4.1.2.3). In order to maintain subatmospheric pressure, 2000 m^3 of air at most is extracted per day from the primary cavities and transferred to the vent stack.

Re b)

The radioactivity entering the exhaust air as a result of leakages from the primary system and connected primary coolant-conveying systems is mainly in the form of noble fission gases.
The radioactivity released as aerosols and iodine isotopes is lower by several orders of magnitude. On account of the low activity of the primary coolant, the quantity of radioactivity released via this path is on the whole very low. The calculations are based on an integral leakage of one per thousand per day from primary coolant-conveying systems, for an average air change rate of 1/h in the affected compartment areas and unfiltered release.

Re c)

The main source of gaseous radioactive waste is the regeneration of the helium purification system. The resulting regeneration gases are collected in the storage tanks for radioactively contaminated helium where they are stored until release via the vent stack. The activity input into the storage tanks for radioactively contaminated helium from other primary coolant-conveying systems via the dump system is negligible in comparison to the activity burden of the regeneration gases.
Most of the activity in these tanks originates from the medium and long-lived radionuclides such as Xe133, Kr85, H3 and C14.
Regeneration of one train of the helium purification system is performed approx. every 1000 h. A period of about 5 weeks is available for each release from the storage tanks for radioactively contaminated helium from the vent stack. This allows quasi-continuous release.
15 regeneration operations per year were assumed in the calculation of the radioactive release rates. This is equivalent to a plant capacity of 7500 equivalent fullpower hours for the HTR Module power plant. The decay of short-lived radionuclides because of quasi-continuous emptying of the storage tanks for radioactively contaminated helium is not considered in the calculation.
On regeneration of the helium purification system, most of the previously separated tritium is transferred in the form of liquid HTO to the tank for condensate collected in operation. However, the input of a small fraction into the storage tanks for radioactively contaminated helium with the regeneration gas cannot be ruled out. This fraction was assumed to be 2%.

Re d)

In comparison to the sources of radioactive releases with exhaust air as discussed above, the activity burden contributed by the gas evacuation system for fuel handling equipment is negligible.
The evacuated air is mixed with the exhaust air flow via HEPA filters in the gas evacuation system.

Re e)

During maintenance and repairs to opened, contaminated systems the exhaust air from the affected HVAC train is switched to the operational exhaust air filter unit because of the possibility of aerosol releases into compartment air. The expected radioactive release is negligible.

1.1.3 Exhaust Air from the Turbine Building

Apart from release of radioactive materials from the vent stack, a small amount of tritium activity is released to the atmosphere by the roof fans of the turbine building. This is the tritium fraction which permeates through the steam generator tubes into the steam/power conversion system and enters the turbine building exhaust air via steam leaks. The calculation is based on a 1:4 distribution of the tritium release from the water/steam cycle to the exhaust air and liquid effluent pathways.

1.1.4 Release of Gaseous and Aerosol Radioactive Materials

Using design data for the various influencing factors, release rates were calculated for the sources of gaseous and aerosol radioactive materials.

On the basis of these calculations, the following release rates are stated in the licence application:

	Bq/a	(Ci/a)	
Noble gases	7.4×10^{13}	(	2000)
Argon 41 fraction	1.2×10^{13}	(approx.	320)
Iodine 131	2.2×10^{7}	(approx.	0.0006)
Long-lived aerosols (Half-life greater than 10 d)	3.7×10^{7}	(	0.001)
Tritium	3.7×10^{12}	(	100)
Fraction released via turbine building roof fans	8.9×10^{11}	(approx.	24)
Carbon 14	8.9×10^{11}	(approx.	24)

1) 3.7×10^{10} Bq is equivalent to 1 Ci

The short-lived comparison aerosols and the iodine isotopes are of small radiological significance.

Solids

Solid waste production

Solid waste production is as follows:

- Evaporator concentrates: approx.4 m^3/a for the first10 years
  approx. 8 m^3 /a in subsequent years
- Decanter residues: approx. 4 drums/a
- Solid operational wastes: approx. 10-40 m^3/a
- Filters, e.g. from the controlled area HVAC systems: approx. 5 m^3/2a
- Wear parts from nuclear systems: approx. 1 m^3/a
- Filter drums from decontamination systems: approx. 4 drums/4a

Conditioning

The solid radioactive waste generated in the power plant is conditioned in such a way as to meet the storage requirements.

Evaporator condensates can be treated in mobile treatment units in the power plant, or the waste can be transported in liquid form to an offsite conditioning plant.

At present, units for immobilization in cement or concentrate drying units are mobile processing facilities (operated by service companies) which are in widespread use.

Evaporator concentrates are solidified in 200 l drums. If conditioned offsite, the drums are shipped back to the plant.

Decanter residues are filled into 200 l drums and are stored pending dispatch to an offsite incineration plant. After offsite conditioning the decanter residue ashes, which are immobilized in cement, are returned to the plant in 200 l drums.

Solid operational waste is collected in containers and sorted into combustible and non-combustible waste. The solid operational waste is then shipped to an offsite conditioning plant and, after conditioning, is returned to the plant in 200 l drums in the form of waste or ash which has been immobilized in cement.

The filters from the HVAC systems of the controlled area are sealed in plastic sheets,and collected in containers. After offsite conditioning, the filter ashes, which have been immobilized in cement, are returned to the plant in 200 l hooped drums.
Wear parts from nuclear systems are filled into 200 l hooped drums and are cast in cement lime in a mobile processing unit.
Filter drums from decontamination systems are dewatered and cast in cement lime in a mobile processing facility.

Storage

After conditioning of the solid waste, about 90 drums (200 l) are to be expected an average per year. These drums are temporarily stored in the drum store in the reactor auxiliary building (storage capacity: approx. 200 drums) until they are transported from the plant to an offsite repository.