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USE OF ION EXCHANGE FOR THE
TREATMENT OF LIQUIDS IN
NUCLEAR POWER PLANTS

K. H. Lin

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ABSTRACT

A study was made to investigate the current and future use of ion exchange (demineralization) as a method for treating liquid radioactive streams at nuclear power plants. Pertinent data were obtained by contacting utility companies, nuclear-steam-supply system vendors, selected AEC-operated facilities, as well as ion exchange resin and equipment manufacturers. Principal emphasis was placed upon obtaining data concerning the decontamination of aqueous solutions characterized by levels of radioactivity that range from 10^{-7} to 1 $\mu\text{Ci/ml}$.

Ion exchange media commonly used in nuclear power plants are synthetic organic resins of polystyrene matrix. They are utilized primarily in the mixed-bed (deep-bed) ion exchange system. Powdered resin (mixed) systems (so-called "filter-demineralizer") are also used in several recent boiling-water-reactor plants. The term decontamination factor (DF), the ratio of the feed to effluent concentration, is widely used and is assumed by designers and operators of the plants to express the ion exchange system performance. In some cases, such DF values may not represent the true system performance. To achieve a desired DF, the feed and effluent must be sampled for the nuclides of interest and the processing discontinued when the desired effluent concentration is exceeded.

Included in this report is a list of average DF values that can be obtained for various ion exchange systems and various groups of radionuclides if good engineering practice is used in the design and operation of these systems. These values are based on ion exchange fundamentals, literature data, laboratory experiments, and plant operating experience. They represent time-average values expected under normal operating conditions rather than maximum values attainable under optimum conditions.

1.0 SUMMARY AND RECOMMENDATIONS

Characterization of various liquid streams in light-water-cooled nuclear power plants in terms of detailed radiochemical compositions is rather limited even though 24 plants are currently (December 31, 1972) in operation. In the primary coolant, the fission products ^{131}I and $^{140}\text{Ba-La}$ and the activated corrosion products ^{51}Cr , ^{58}Co , and ^{60}Co are among the radionuclides having relatively long half-lives (>8 days) and high concentrations ($>10^{-4}$ $\mu\text{Ci/ml}$ each). Tritium and dissolved noble gases are not considered since ion exchange does not materially affect either. The results of this survey on the application of ion exchange technology to the treatment of liquid radioactive streams at light-water-cooled nuclear power plants show that:

- (1) Resin types used most frequently by nuclear power plants are strong-acid cation and strong-base anion exchange resins of polystyrene matrix. These resins are routinely used in many large-scale water treatment operations, and the results of much practical experience are available.
- (2) Mixed-bed units are the most widely employed ion exchange systems, although separate cation and anion units are in common use. Conventional systems use resins with a particle size range of 0.4 to 0.7 mm in beds that vary from 2 to 8 ft in diameter, from 3 to 6 ft in depth, and contain from 30 to 180 ft³ of resin. In a number of boiling-water-reactor (BWR) plants, a powdered form of resin (over 90% of the particles are smaller than 0.03 mm) is used in very shallow beds (8 to 30 ft³ of resin) with large surface area. Large powdered-resin units may be 6 ft in diameter, 6.5 ft in height, contain many thin layers (1/8 to 1/2 in. thick), and have a total surface area of about 800 ft². They function as a filter as well as a demineralizer and are used to treat primary coolant and steam condensate.
- (3) Separate-bed units are used to treat steam generator blowdown, primary coolant, and feed and condensate in the boron recycle system at pressurized-water-reactor (PWR) plants.

- (4) Flow rate of feed to various ion exchange systems ranges from approximately 1 to 50 gpm per square foot of resin bed cross-sectional area for BWR plants, the highest rate being applicable to steam condensate. In PWR plants, the variation is not as wide and generally falls in the range 5 to 20 gpm/ft²; the higher rates are frequently used for the letdown stream from the primary coolant system. The flow rate in the powdered-resin systems is about 4 gpm per square foot of resin bed surface area.
- (5) Temperatures of the feed streams are usually below 140°F. Higher temperatures can cause increased deterioration of resins and shorten their service life.
- (6) The pH values of the feed streams in BWR plants are near neutral under normal conditions, but in PWR plants the pH values may range from 4 to 10.5.
- (7) Performance data on ion exchange systems with respect to specific radionuclides are not readily available in spite of the important role played by these systems in nuclear power plants and other types of nuclear installations. Most data are based on gross ($\beta + \gamma$) radioactivity measurements which do not distinguish between the various chemical and physical species in the feed and effluent. Electrical conductivity of the effluent and pressure drop across the resin bed are other properties frequently monitored at BWR plants. Many plants do measure radioiodine. The determination of other individual radionuclides is strongly recommended.
- (8) Decontamination factor (DF) is defined as the ratio of the feed to effluent concentration. Although DF is used by designers and operators to express the performance of an ion exchange system, it is not generally the determining factor for resin replacement or regeneration. Such replacements or regenerations are often scheduled for convenience in plant operation (during refueling), are determined by radiation levels around the ion exchange equipment, or are dictated by excessive pressure drop across the bed.

- (9) Radionuclides exist in different forms which have different removal characteristics. Some exist primarily as ionic species which behave in accord with ion exchange principles. Some exist as insoluble particles of various sizes, and their behavior is related to particle properties in regard to filtration and adsorption on surfaces. Some tracer level ions adsorb on (or exchange with ions in) these solid particles, and subsequently behave like the solids. As a result, some nuclides do not behave as their chemical properties might indicate. Their behavior may be different for different streams within a plant and from plant to plant. Thus, removal of particulates may be as important as removal of ionic constituents.
- (10) Modes of operation of ion exchange systems often differ from that in water treatment experience. In particular, extremely large specific volumes of water may be processed before resin regeneration or replacement. Superior performance could be achieved by more frequent resin replacement or more complete and frequent resin regeneration.
- (11) Recycling of effluent from an ion exchange system through the same system to improve the quality of the effluent will do no good if the resin is loaded or partially loaded, if the distribution coefficient is low for the species being removed, or if the species is in a nonexchangeable form. Resin replacement (or regeneration) or a different resin (or process) is needed for improved performance in such cases.

The results of laboratory experiments at Oak Ridge National Laboratory (ORNL) using feed solutions containing trace amounts of ^{137}Cs , ^{131}I , and Sr, and large amounts either of NaCl (1000 and 2500 ppm) or of a mixture of Na_2HPO_4 (1000 ppm), H_3BO_3 (1700 ppm), and LiOH (7 ppm) bring out that: (1) Breakthrough of trace radionuclides as well as nonradioactive nuclides takes place earlier when the feed flow rate or the concentrations of the nonradioactive nuclides are increased. (2) Empirical rules for ionic selectivity for ion exchange resins (strong acid and

strong base) hold for the solutions studied. (3) DF's can be less than 1, i.e., concentrations of nonradioactive ions, Li^+ , PO_4^{3-} , and BO_3^{3-} , in portions of the effluent can exceed the concentrations in the feed. Briefly stated, these results show that these solutions, at trace levels of radioactivity, behave as would be expected from ion exchange theory and experience.

Table 1 lists average DFs that can be attained for various ion exchange systems and various groups of radionuclides in routine plant operations. These DFs represent time-average (or volume-average) values over the operating life of the column. They are derived from water treatment experience, ion exchange fundamentals, laboratory data, and plant operating experience. They take into consideration anticipated operational occurrences. They are not intended as upper limits or as values typical of unusually careful operation and should be reexamined as new data are obtained for specific nuclides in operating systems. The DFs in Table 1 are considered to be attainable in the treatment of aqueous solutions having concentrations of radioactivity that are less than $0.1 \mu\text{Ci/ml}$, provided that a sufficient amount of ion exchange resin (e.g., strong acid and strong base, polystyrene matrix) is employed, that concentrations of both suspended and dissolved solids are normally less than 1000 ppm, and that good engineering practice is used in the design and operation of these systems. These values are justified by plant operating experience supported by laboratory data.

In summary, the DFs listed in Table 1 represent average values expected over extended operation rather than maximum values indicating DFs attainable under optimum conditions. For a given ion exchange system, the DFs will be higher if less waste is processed before the ion exchange bed is replaced or regenerated (and will be lower if more waste is processed). However, each ion exchange system can attain a predicted or desired DF, if the feed and effluent are measured frequently, and if the feed is terminated and the resin replaced or regenerated when the maximum permissible concentration in the effluent is reached. The rationale for these values is given in the following sections.

Table 1. Decontamination Factors That Can Be Attained with Various Ion Exchange Systems

	Decontamination Factor (Average Over the System Life-Time)		
	I	Cs, Rb	All Other Solubles
	<u>PWR</u>		
Mixed bed ($\text{Li}^+ - \text{BO}_3^{3-}$ form)	10	2	10
Cation bed (H^+ form)	1	10	10^2 (cations)
Mixed bed ($\text{H}^+ - \text{OH}^-$ form) (radioactive waste)	10^2	2	10^2
Anion bed (OH^- form)	10^2	1	10^2 (anions)
	<u>BWR</u>		
Mixed bed ($\text{H}^+ - \text{OH}^-$ form)			
(a) Primary coolant	10	10	10
(b) Steam condensate	10^3	10	10^3
(c) Clean waste	10^2	10	10^2
(d) Dirty waste	10^2	2	10^2
Cation bed (H^+ form) (dirty waste)	1	10	10^2 (cations)
Powdex in any system	10	1	10
	<u>PWR and BWR</u>		
Mixed bed ($\text{H}^+ - \text{OH}^-$ form) (evaporator condensate)	10	10	10

1.1 Decontamination Factor (DF) Values for PWR Plants

In PWR plants relatively large amounts of added chemicals are present in both the primary and secondary coolants (e.g., H_3BO_3 and LiOH in primary coolant and Na_2HPO_4 in secondary coolant). Accordingly, even if the same DF is assigned to comparable ion exchange systems in both PWR and BWR plants, the service life of the system in the PWR plant may be shorter than that in the BWR plant, since the added chemicals compete for the exchange sites and cause the trace nuclides to appear in the effluent sooner than if only the trace nuclides were present in the feed. In other words, more competing ions cause a lower distribution coefficient which causes a higher effluent concentration, and lower DF, and earlier breakthrough.

The use of a lower DF ($\text{DF} = 2$) for cesium and rubidium in mixed-bed systems (both $\text{Li}^+-\text{BO}_3^{3-}$ and H^+-OH^- forms) than those in a cation bed (H^+ form) ($\text{DF} = 10$) for the PWR is based on the assumption that appreciable amounts of competing ions are present in the feed to the mixed-bed systems. Ion exchange resins will preferentially remove polyvalent metallic nuclides over monovalent cesium. Therefore, these nuclides are assigned a higher DF than cesium.

1.2 Decontamination Factor (DF) Values for BWR Plants

In BWR plants the mixed-bed ion exchange system for the steam condensate is assigned a higher DF ($\text{DF} = 10^3$) than a mixed-bed ion exchange system for primary coolant. This assumption is supported by plant operating data. The steam condensate generally contains a different distribution of radionuclides (mainly activated corrosion products) than the primary coolant.

A lower DF is assumed for liquid radioactive waste ($\text{DF} = 10^2$ for I and ions other than Cs and Rb) compared to that for the condensate because of the presence of an appreciable amount of soluble and insoluble solids in the waste, which tends to reduce the capacity of the ion exchange

resin bed for removal of radionuclides. A lower DF value ($DF = 10$) for Cs and Rb is based on the assumption that the feed to the ion exchange units contains significant amounts of competing polyvalent ions.

2.0 BACKGROUND

Various existing techniques in chemical engineering operations have been adapted for decontamination of plant liquid waste streams to reduce the radioactive nuclides that will be released to bodies of water under controlled and monitored conditions in accordance with federal regulations.^{1,2} Among the techniques employed for the treatment of liquid streams are ion exchange (demineralization), filtration, evaporation, scavenging, and precipitation. The ion exchange technique has been shown to be an effective method for the removal of trace quantities of radionuclides from nuclear power plant liquid wastes. A survey on the use of evaporation for the treatment of liquid radioactive streams is given in ref. 3.

Although uses of the ion exchange technique in nuclear power plants are extensive, plant data on the performance of the ion exchange system are quite limited. In addition, the method used to measure the performance of the equipment varies from one plant to another. Likewise, the performance data may change with the plant design parameters. These variables present difficulties, not only in the evaluation of the effectiveness of the ion exchange system in removing various radionuclides but also in comparing equipment performance at different plants.

The objective of this study was to prepare a generic report that presents information pertaining to the performance and use of ion exchange systems in the treatment of liquid radioactive wastes at a number of nuclear power plants in the United States. Also included in the study are selected AEC-operated facilities where data on the performance of ion exchange systems with respect to specific nuclides were available. The study covers an investigation of the current and future use of ion exchange in various nuclear installations.

As a part of this study, a number of installations were contacted to obtain performance data on various ion exchange systems. They included 13 utility companies with operating nuclear power plants, 14 demineralizer and ion exchange resin manufacturers, 4 nuclear-steam-supply-system vendors, and 2 architect-engineers. In addition, data were gathered from selected sites at the National Reactor Testing Station (NRTS), Idaho. Review of the published literature emphasized the treatment of aqueous solutions characterized by concentrations of radioactivity in the range of 10^{-7} to 1 $\mu\text{Ci/ml}$. Fundamentals of ion exchange were considered to facilitate the technical assessment of the data gathered in the study. In addition, a series of laboratory-scale experiments were carried out at Oak Ridge National Laboratory (ORNL) with radioactive liquids simulating streams at nuclear power plants to support the data gathered in the survey.

3.0 SOURCES AND CHARACTERISTICS OF LIQUID RADIOACTIVE STREAMS IN NUCLEAR POWER PLANTS

The light-water-cooled boiling-water reactor and pressurized-water reactor represent the two major classes of nuclear power reactors. They are basically different with regard to the method of generation of steam that drives the steam turbine for power production. In a typical BWR, the cooling water in contact with nuclear fuel elements is allowed to boil under a pressure of 1000 to 1500 psia. Steam thus produced is transported directly to the turbine, and the steam condensate is returned to the reactor for reuse (Fig. 1). In a typical PWR, the primary coolant under a pressure of ~ 2200 psia flows through an external steam generator and is subsequently returned to the reactor for reuse (primary coolant, Fig. 2). A separate secondary coolant stream acquires heat from the primary coolant through the heat transfer surfaces in the steam generator (Fig. 2). Thus, the primary coolant and the secondary coolant should never directly contact each other unless a leak in the steam generator occurs. Steam generated in the secondary system flows to the turbine, and the condensate becomes

Fig. 1. Typical System for Treating Liquid Streams in a Boiling Water Reactor Nuclear Power Plant.

ORNL DWG. 72-11734

Fig. 2. Typical System for Treating Liquid Streams in a Pressurized-Water Reactor Nuclear Power Plant.

the major portion of the feedwater to the steam generator. The reactor coolant in the primary system normally contains added hydrogen and hydrazine for oxygen scavenging, enriched $^7\text{LiOH}$ for pH control to reduce corrosion, and H_3BO_3 for control of the core reactivity. Occasionally, Na_2HPO_4 is added to the secondary coolant to make the water slightly alkaline.

A simplified schematic flow diagram of a typical liquid radioactive waste treatment system for a BWR power plant is shown in Fig. 1. The specific scheme used varies somewhat from one plant to another. The primary coolant, steam condensate from the turbine, and liquid radioactive waste are the three major liquid streams that are generally amenable to treatment by ion exchange. Also shown is the ion exchange system for the fuel storage pool water. The radioactive contaminants in the condensate consist mainly of: corrosion products (Fe, Co, Mn, and Cr) that originate in the turbine, condenser, and reactor core and become activated as the coolant is circulated through the reactor; and fission products from any leaking fuel elements.⁴⁻⁷ Liquid radioactive wastes are segregated into four types (high purity, low purity, chemical, and detergent) according to their electrical conductivity and chemical properties as indicated in Fig. 1.

A simplified schematic flow diagram of a typical liquid radioactive waste treatment system for a PWR power plant is shown in Fig. 2. The major liquid streams amenable to treatment by ion exchange are the primary coolant, the secondary coolant, and liquid radioactive waste. As with BWRs, fission products from the leaking fuel elements and activated corrosion products originating in the primary system are the major sources of radioactivity in the primary coolant.^{8,9} In contrast to the BWRs, this primary coolant may contain appreciable concentrations of dissolved ionic solids. Purification of the coolant is accomplished by circulating a side stream (letdown) from the main coolant flow through the reactor coolant treatment system (which includes the chemical and volume control system and the boron recycle system) containing filters and ion exchange units to remove suspended and dissolved contaminants. Leaks in the steam

generator result in transfer of radionuclides to the secondary coolant. A portion of this water is removed continuously (blowdown) to maintain the coolant chemistry. The blowdown stream, which may have its own treatment system, is processed by ion exchange or other methods prior to discharge or reuse. An additional source of liquid radioactive waste in a PWR is the system for adjusting the boron concentration of the coolant. The liquid wastes can be classified into four categories (high purity, low purity, chemical, and detergent) as indicated in Fig. 2.

4.0 ELEMENTS OF ION EXCHANGE TECHNOLOGY

A brief review of some of the basic principles of ion exchange is presented below to facilitate the interpretation of operating data to be discussed in later sections. More detailed supplementary information is given in Appendix D and in standard reference books on ion exchange.¹⁰⁻¹⁵

Many materials found in nature (humus, silicates, phosphates, and resins) possess variable ion exchange capacities. Synthetic inorganic ion exchangers are relatively new, but their development is progressing rapidly. Of the synthetic ion exchangers, the organic type is most used. A wide variety of natural and synthetic ion exchange materials are commercially available.

Ion exchange is a process in which a reversible stoichiometric interchange of ions of the same sign takes place between an electrolyte solution and a resin or exchanger as illustrated in Eqs. (D-1) and (D-2) in Appendix D. The exchange involves no permanent alteration in the structure of the resin. The very nature of the ion exchange process demands that radionuclides in various liquid streams at power reactor plants be in ionic form for this decontamination process to be effective. Nonionic forms [insoluble particulates, colloids, as well as neutral molecules (I_2) and neutral complexes] which are incapable of undergoing ion exchange can be responsible for poor solution decontamination. Varying fractions of some particulates and colloids may pass through the resin bed, and the part which is trapped may coat the resin particles and reduce their efficiency.

4.1 Types of Ion Exchange Resins Used in Nuclear Power Plants

Styrene-divinylbenzene copolymers are the network materials used in the preparation of synthetic ion exchange resins widely employed in nuclear power plants. Table 2 lists the characteristics of representative strong and weak acid and base types as determined by the active functional groups. Ionic forms other than H^+ and OH^- (Li^+ , K^+ , NH_4^+ , and borate form) are also available for specific applications.

Strong-acid cation resins (H^+ form) are characterized by their ability to readily interact with a weakly ionized or neutral salt, dissociating the salt into its acid in solution and converting the resin to the salt form. Likewise, a strong-base anion resin (OH^- form) will dissociate a weakly ionized or neutral salt into its base in solution, simultaneously converting the resin to the salt. A disadvantage common to these resins is low regeneration efficiency, i.e., they require a large excess of regenerants in relatively high concentrations. In contrast, a complete regeneration of both weak-acid cation and weak-base anion resins is achievable with nearly stoichiometric amounts of regenerants. However, they have disadvantages in that they are slow and less effective in dissociating weakly ionized and neutral salts. In addition, the ion exchange efficiency of weak resins is considerably more sensitive to the solution pH than that of strong resins. Further, the weak resins are generally more expensive and have a shorter life expectancy than the strong resins. Nevertheless, the smaller amounts of regenerants required for weak resins may be an important advantage in view of possible problems associated with disposal of regenerants.

The quality of water being used in light-water-cooled nuclear power reactors must be very high in order to minimize corrosion problems, and to reduce formation of neutron activation products and radiochemically produced compounds. To satisfy this requirement, nuclear-grade ion exchange resins¹⁶ were introduced commercially. They are basically the same as the regular grade with regard to structure, but have much smaller amounts of leachable impurities (organics in particular); more uniform particle sizes (cation resins: ~ 0.45 mm; anion resins: ~ 0.6 mm, $< 1\%$ fines); and a higher degree of conversion to the desired ionic form ($> 95\%$ for cation resins, $> 80\%$ for anion resins).

Table 2. Characteristics of Typical Ion Exchange Resins

Resin Type	Functional Groups	Typical Regenerant	Regeneration Level (%)	Theoretical Capacity (meq/ml wet resin)	Ionic Form	pH Range	Max. Oper. Temp. (°F)
Strong acid	Sulfonic	4% H_2SO_4	95	1.5-2.0	H^+	1-14	250
Strong base Type 1	Quaternary ammonium Type 1	4% NaOH	80	0.5-1.0	OH^-	1-14	140
Strong base Type 2	Quaternary ammonium Type 2	4% NaOH	80	0.5-1.0	OH^-	1-14	100
Weak base	Primary, secondary, and/or tertiary amine	4% NH_4OH	95	1.5-2.5	OH^-	1-7	200

4.2 Factors Affecting Performance of an Ion Exchange System

The performance of an ion exchange system is governed by the physical and chemical properties of the exchange resin as well as by the operating conditions. Important properties of the ion exchange resin include (1) exchange capacity, (2) swelling equilibrium, (3) degree of cross-linking, (4) resin particle sizes, (5) ionic selectivity, (6) ion exchange kinetics, and (7) chemical and physical stability. Important operating conditions include (1) chemical and physical forms of species in feed solution (ionized or nonionized species, ionic sizes, and valences); (2) temperature, pH, and flow rate of feed solution; (3) design and dimension of ion exchange beds (method of packing, whether a multistage system with separate cation and anion exchanger units or a mixed-bed system is used; height-to-diameter ratio); and (4) level, or degree of completeness, of regeneration.

Although properties of ion exchange resins and operating conditions are discussed individually below and in Appendix D, they are interrelated so closely that they must be considered in toto to properly evaluate an ion exchange system.

4.2.1 Ionic Selectivity

Numerous theories have been developed to explain factors influencing ionic selectivity. However, no satisfactory quantitative prediction of this property can be made for a system involving a multicomponent ionic mixture. Current practice in the use of ion exchange systems is to estimate the selectivity based on the empirical rules listed in Table D-1 of Appendix D. These rules are usually applicable to dilute aqueous solutions at ordinary temperatures. The selectivity tends to diminish with increasing concentration and temperature. Exceptions to the rules are also found.

4.2.2 Particle Size

Particle sizes of regular-grade strong-acid cation and strong-base anion resins commonly available on the market usually range from approximately 0.4 to 0.7 mm. The effect of particle size on the diffusion of ions across the film or layer of solution surrounding a resin bead and on the diffusion of ions within a resin bead is illustrated by Eqs. (D-20) through (D-22) in Appendix D.

The use of small particle size resin will increase the number of theoretical plates¹⁵ in a given column; however, the pressure drop across the ion exchange column will be greater for the same flow rate.

Consideration of the advantage of small particle sizes has led to the development of a fine powdered mixture of cation and anion ion exchange resins (>90% -325 mesh) which is manufactured by the Graver Water Conditioning Company under the trade name "Powdex." Use of such powdered resins involves coating the outside surfaces of filter cartridge elements (nylon; 2.5 in. diam $\times$ 10 in. high; ~ 0.5 ft² surface area) with thin layers (1/8 to 1/2 in.) of resin.¹⁷ In these arrangements, the resin functions both as a filter and as an ion exchanger. Four or more cartridges are stacked in tiers, one above the other, and the entire assembly is placed in a pressure vessel. A large pressure vessel unit may be 6 ft in diameter and 6.5 ft in height and contain as many as 300 filter cartridges with a total surface area of about 800 ft². With a feed flow of about 4 gpm/ft², the pressure drop may range from approximately 2 to 5 psi for fresh resins and from 20 to 30 psi for resins near exhaustion.

Exhausted powdered resins are not regenerated. Thus, the time and cost associated with regeneration are eliminated. However, the precoating step usually requires 1 to 1-1/2 hr. Powdered ion exchange resins are used in other industries for the efficient removal of colloidal silica, iron, copper, and nickel. The equipment is simple and the pressure drop across the bed is relatively low when the resin is fresh. An economic evaluation of the powdered resin system applied to BWR plants is presented in ref. 18. Some of the disadvantages are relatively high initial cost and relatively high replacement costs of filter cartridges.

4.2.3 Mixed-Bed vs Separate-Bed System

The ion exchange system most frequently employed in water and liquid waste treatment is the mixed-bed system which consists of a stationary bed containing mixed anion and cation resins (Fig. 3). The liquid stream is fed from the top of the bed, and percolates downward through the bed during operation of the process. When ionic contaminants begin to appear in the

(a) SEPARATE BED SYSTEM

(b) MIXED-BED SYSTEM

Fig. 3. Schematic Diagram of Mixed-Bed and Separate-Bed Ion Exchange Systems.

effluent in significant quantities (breakthrough), the bed is considered exhausted or loaded and the feed is discontinued. The bed is then regenerated (followed by rinsing with deionized water) or the bed is discarded and replaced with fresh resins. These beds containing resins with a particle size range of 0.4 to 0.7 mm may have dimensions varying from 2 to 8 ft in diameter and from 3 to 6 ft in depth and contain from 30 to 180 ft³ of resin. The flow rates through these units range from 1 to 50 gpm per square foot of resin bed cross-sectional area, and the pressure drop may reach 25 psi. Pressure drops as well as volumes of feed processed per volume of resin are fairly similar to those for the powdered-resin systems (see Sect. 4.2.2 and Table G-1 in Appendix G).

A disadvantage with the mixed-bed system is the time required for regeneration, which entails separation of the two types of resin by flotation making use of the difference in density, followed by transfer to individual regeneration vessels. After regeneration, the resins are again thoroughly mixed and returned to the column. Shown in Fig. 3b are separated cation and anion resin layers, representing the condition for regeneration. Some mixed-bed systems are designed to permit regeneration in a single vessel.¹¹ An advantage of the mixed-bed system is that the cation and anion exchange reactions take place simultaneously and ions removed from the resin (e.g., H⁺ and OH⁻) react with each other, thereby reducing the electrolyte concentration. The favorable ionic equilibria thus achieved in the column result in a high column efficiency.

For removal of both cation and anion contaminants from liquid waste, two separate ion exchange beds in series, a cation bed ahead of an anion bed, are sometimes used instead of a mixed-bed system. The separate bed arrangement may be preferred for the purpose of removing specific radionuclides. Regeneration is much simpler with a separate-bed than with a mixed-bed unit.

4.3 Decontamination Factor

The degree of column utilization, defined as the ratio of the breakthrough capacity to the total capacity, is often employed to express the

efficiency with which the ion exchange column has been used. The steeper the breakthrough curve (a plot of time or the cumulative effluent volume vs the ratio of the concentration of a component in the effluent, C_e , to the concentration of the same component in the feed, C_f), the higher is the degree of column utilization. This is illustrated qualitatively in Fig. D-4 of Appendix D. The reciprocal of the fractional breakthrough is defined as the instantaneous DF. In other words, the instantaneous DF, $(DF)_i$, is given by the expression

$$(DF)_i = C_f / C_e . \quad (1)$$

Speaking broadly, $(DF)_i$ is high early in the life of a bed and becomes lower as breakthrough is approached. The ratio of the total activity of a component in the cumulative feed, A_f , to the total activity of the component in the cumulative effluent, A_e , is defined as the time-(or volume-) average DF. The activities are considered normalized to a specified decay time. In other words, the time-average DF is given by the expression

$$DF = A_f / A_e . \quad (2)$$

The DFs thus defined are used to characterize the performance of an ion exchange system and are governed by the operating conditions as well as the properties of the ion exchange material.

5.0 USE OF ION EXCHANGE AT NUCLEAR POWER PLANTS

Under normal conditions, the concentrations of individual radionuclides in liquid streams at nuclear power plants are much less than parts per billion. As shown by the compilation of results for nuclear research sites in ref. 19 and for the various nuclear installations included in the list of publications given in Appendix E, a well-designed and operated ion exchange system can be quite effective in reducing such trace quantities of radionuclides. The principal emphasis in this survey was placed upon obtaining data concerning the decontamination factors achieved in ion exchange systems used at nuclear power plants. A limited number of other

types of nuclear installations were surveyed. These include the Test Reactor Area (TRA) at the National Reactor Testing Station (NRTS), Idaho, and the Idaho Chemical Processing Plant (ICPP), Idaho. Brief descriptions of the installations, as well as characteristics of their liquid radioactive wastes, are presented in Appendix B for the TRA and Appendix C for the ICPP.

The data obtained (through December 31, 1972) from the organizations that responded (Appendix F) to the inquiries are summarized in Table 3, Appendices A-C, G, and H. Much of the information received contained little detail on influent and effluent stream compositions, operating conditions, decontamination factors, or problems and the actions taken to correct them. A supplemental survey and compilation will prove valuable as the nuclear industry gains more experience and operating information. The data obtained in this survey are discussed in the sections that follow.

Table G-1 in Appendix G summarizes pertinent data for ion exchange systems in six BWR plants currently in operation and five plants that are under construction. The list represents plants with installed power ranging from 240 to 3294 MW(t) and covers equipment specifications as well as some performance data. Table H-1 in Appendix H summarizes pertinent equipment specifications and performance data for ion exchange systems in 11 PWR power plants (Indian Point 1 and 2 were considered as separate plants; 7 operating plants, 4 under construction). The installed power for these plants ranges from 505 to 3423 MW(t). The information (feed conditions, effluent monitoring, and DF) for the BWR and PWR plants under construction consists of the estimated values and assumptions that have been used as the design basis by utility companies.

5.1 Radiochemical Composition of Feed

Not all radionuclides are in solution in the liquid radioactive streams at nuclear power plants. Transition elements and rare earths tend to form precipitates while others form ionic species. Some metallic transition elements form compounds in solution as anions rather than cations (^{51}Cr and ^{99}Mo). The probable chemical and physical forms of various radionuclides

Table 3. Summary of DF Values Used by Operators and Designers of Nuclear Power Plants

Type of Ion Exchange System	Liquid Stream Treated	DF (Average Over Life-Time of Resin)	Physical Form of Nuclides
<u>BWR</u>			
Mixed bed	Primary coolant	10	Soluble plus insoluble
Powdex	Primary coolant	10	Soluble plus insoluble
Mixed bed	Steam condensate	100 3-5	Soluble Insoluble
Powdex	Steam condensate	50-100 10-20	Soluble Insoluble
Mixed bed	Radioactive waste	1000 5	Soluble Insoluble
<u>PWR</u>			
		DF (Average Over Life-Time of Resin)	<u>Nuclides</u>
Mixed bed ($\text{Li}^+ - \text{OH}^-$ forms)	Primary coolant	10	All except Cs, Y, and Mo
		1	Cs, Y, and Mo
Cation bed (H^+ form)	Primary coolant	10	Cs, Y, and Mo
Mixed bed ($\text{H}^+ - \text{OH}^-$ form)	Boron recycle evaporator feed	10	All except Cs, Y, and Mo
		1	Cs, Y, and Mo
Anion bed (OH^- form)	Boron recycle evaporator condensate	10	I, Br, Cl, F
Mixed bed ($\text{H}^+ - \text{OH}^-$ forms)	Radioactive waste	10	Gross $\beta + \gamma$

indicated in Table A-1 (fourth column) are based on operating data as well as on fundamental chemical data for individual nuclides.

Nine BWR plants reported detailed radiochemical compositions of feed solutions to the ion exchange systems. These are summarized in Tables A-1 through A-4 in Appendix A. The radiochemical compositions of samples of primary coolant at the Dresden 1 and Big Rock Point plants are given in Table A-1. Additional examples of radiochemical compositions of primary coolant at BWR plants (Oyster Creek, Millstone Point, Monticello, Dresden 2, and Dresden 3) are shown in Table A-2. Concentration values for individual nuclides vary during the fuel cycle, depending upon plant conditions that are not always predictable. Accordingly, these values should not be considered to represent all BWR plants. The same statement is applicable to the other tables in Appendix A.

Table A-3 illustrates compositions of selected samples of low purity wastes from two BWR plants. A comparison of Table A-3 with Table A-1 reveals that both the number of radionuclides and the total radioactivities in the wastes are much smaller than in the primary coolant. Concentrations of corresponding major radionuclides (Cs, Sr, Fe, Mn, and Co), however, are about the same. Table A-4 shows compositions of samples of fuel pool water and laundry wastes taken at the Dresden 1 plant. The fission products listed represent the predominant radionuclides in the fuel storage pool, but a significant concentration of cobalt isotopes is also present.

Most BWR plants analyze for selected radionuclides in specific liquid streams. For example, one BWR plant may find ^{65}Zn originating from materials of construction to be the predominant nuclide while other plants may find fission product $^{140}\text{Ba-La}$ as the major contribution to the radioactivity. Still another plant may consider removal of activated corrosion products such as ^{60}Co and ^{54}Mn in the crud to be the controlling factor. Radioactivity from iodine isotopes (^{131}I , ^{133}I , and ^{135}I) is of importance to most plants and the amount of iodine is usually determined.

Tables A-5 and A-6 in Appendix A show the information obtained on radiochemical compositions of feed solutions to the ion exchange systems at PWR plants. The concentrations of only a few radionuclides are determined by the majority of PWR plants surveyed, the selection being governed

by a specific operating problem. Iodine isotopes are always found in the primary coolant. The presence of appreciable amounts of cesium was reported by two plants. Cobalt and chromium radionuclides are among the major radioactive species resulting from the activation of corrosion products. Tritium accounts for most of the radioactivity in all liquid streams.

Sample compositions of primary coolant in three different PWR plants are summarized in Table A-5. One is an operating plant, Yankee-Rowe, whereas the other two are hypothetical plants. The major sources of radioactivity (other than tritium) in the primary coolant are neutron activation products, particularly ^{24}Na , and those from materials of process equipment and piping (^{51}Cr , ^{54}Mn , ^{55}Fe , ^{58}Co , ^{64}Cu , and ^{187}W). The radiochemical compositions of samples from the secondary-loop water and liquid wastes of Yankee-Rowe shown in Table A-6 provide evidence of leakage from the primary system because radionuclide concentrations in both the blow-down stream and the turbine condensate are one one-hundredth to one one-thousandth those in the primary coolant.

5.2 Types and Functions of Ion Exchange Systems

Strong-acid cation (H^+ form) resins and strong-base anion (OH^- form) resins, both composed of styrene-divinylbenzene copolymers, are the two types of resins used in BWR plants listed in Table G-1. They may be used in separate-bed, mixed-bed, or powdered-resin systems. The mixed-bed system is, by far, the most widely used. The separate-bed system was not utilized in any of the BWR plants surveyed. Ion exchange systems are used chiefly for treatment of primary coolant, steam condensate, fuel storage pool water, and liquid radioactive waste. The pH values of these liquid streams are approximately neutral, and the temperatures are normally below 140°F . The powdered-resin system is used primarily in the treatment of primary coolant and steam condensate. This type of system is not used in the treatment of radioactive waste in any of these plants, because of the short life of the bed which results from the small amount of resin used in the system as compared with the conventional mixed-bed system.

Although strong-acid cation (H^+ form) and strong-base anion (OH^- form) resins are still the types most frequently used in PWR plants, weak-acid cation resins (Zion plant, Table H-1) and intermediate-base anion resins (Palisades plant, Table H-1) are used in liquid radioactive waste treatment systems. No application of the powdered-resin system at PWRs has been reported. One of the distinct features of ion exchange systems in PWRs is the utilization of the $^7Li^+$ form strong-acid resins (more than 90% of exchange sites in 7Li form) in the mixed-bed unit for the primary coolant to minimize formation of tritium by the $^6Li(n,\alpha)^3H$ reaction. The main purpose for the use of lithium in the coolant is to control the pH.

Use of separate cation and anion exchange systems is quite common in PWRs. Cation units with strong-acid resin in the H^+ form are used mainly to remove lithium and cesium from the primary coolant (letdown stream) and in the boron recycle system. Anion units are used mainly to remove boron from the primary coolant toward the end of the fuel cycle; however, they are also utilized in the removal of boron carryover in the condensate from the boron recycle evaporator. The complexity of the liquid stream treatment system in a PWR plant is reflected in the variety of uses of ion exchange units (Table H-1; also see Fig. 2). Although none of the operating PWR plants included in the survey had an ion exchange system to treat the steam generator blowdown, a few new plants currently under construction have proposed such systems (Zion and Diablo Canyon).

Temperatures of PWR liquid streams are about the same as those of BWR liquid streams ($<140^\circ F$). The pH, however, varies over a wide range (4 to 10.5), especially in the primary coolant because variable amounts of chemicals ($LiOH$, H_3BO_3 , hydrazine, etc.) are present. Changes in pH can strongly influence the chemistry of some nuclides, especially their solubility. Also notable (Table H-1 in Appendix H) is the variation in the flow rate of the letdown stream to ion exchange systems for the primary coolant at PWRs. The flow rate itself is generally much lower than that of the cleanup stream of the primary coolant in a BWR (Table G-1 in Appendix G); therefore, smaller ion exchange units are required.

5.3 Criterion for Regenerating or Replacing Ion Exchange Resins

Determination of the gross ($\beta+\gamma$) radioactivity of feed and effluent is the most common method employed by BWR plants to measure and evaluate the performance of an ion exchange system. Electrical conductivity is also used either alone or in combination with the gross radioactivity as a guide to determine when to replace or regenerate the ion exchange resins. The pressure drop across the ion exchange bed is usually monitored in the primary coolant and condensate systems. When the pressure drop exceeds a preset value, the bed may be cleaned by the ultrasonic technique²⁰ or by backwashing several times to remove the accumulation of crud.

Powdered-resin units are not regenerated; however, regeneration of mixed-bed units for treatment of steam condensate is accepted practice at BWR plants. In certain plants (Dresden and Humboldt Bay), the replacement of the ion exchange resins (for primary coolant and condensate) generally coincides with the refueling schedules. Some plants replace their resins when the radioactivity of the ion exchange unit reaches a level that could interfere with the accessibility of the equipment or the handling of the resins in the waste system. Thus, the ion exchange system performance is not always the factor in determining when resins should be replaced or regenerated. As a result, the average service life between regenerations or replacements ranges from about a week for radioactive waste to one year for primary coolant. Approximately one to two months is the average service life for condensate ion exchange systems. The life is extended when the system is cleaned by the ultrasonic technique on a regular basis.

As in BWR plants, gross ($\beta+\gamma$) radioactivity is the principal parameter used by PWR plants to evaluate the performance of an ion exchange system. Electrical conductivity is rarely used as the parameter for monitoring primary coolant, but its application to monitoring of the ion exchange system for the evaporator condensate has been utilized by a few plants. In general, evaluation of the effectiveness of removal with respect to specific nuclides is sporadic.

Unlike BWR plants, ion exchange systems in PWR plants are normally not regenerated. Exceptions include the condensate from once-through steam generators, ion exchange units used to remove boron from primary coolant, and some anion exchange units used to treat condensate from the boron recycle evaporator. Replacement of resins is largely based on the gross ($\beta+\gamma$) DF. Several plants are guided by the DF values for iodine, lithium, and cesium in the primary coolant and other liquid streams. The level of radiation from the radioactivity accumulated on the resins is another factor used.

The Yankee-Rowe plant, which has three different ion exchange units (one each of the mixed-bed, cation, and anion units) for the primary coolant, employs a flexible mode of operation in which only one unit is used at a time. The mixed-bed unit is used most of the time. Variables monitored include gross radioactivity, specific concentration of major radioactive nuclides, electrical conductivity, boron, Ni_4^+ , and Ci^- . Occasionally the chlorine content (<0.05 ppm) becomes the controlling factor for replacement of the ion exchange unit. When the primary coolant is borated, $\text{H}^+-\text{BO}_3^{3-}$ forms of resins are used, while $\text{NH}_4^+-\text{OH}^-$ forms are utilized when operating in the alkaline range with approximately 12 ppm of NH_3 in solution. At a pH near 7, H^+-OH^- form resins are used.

5.4 Decontamination Factor

The majority of BWR plants use a DF based on gross ($\beta+\gamma$) radioactivity to represent system performance. The limited analyses of feeds and effluents for specific nuclides that have been reported by some plants are largely based on "grab" samples and are not amenable to a systematic evaluation of the ion exchange units. The DF values generally used by the operators and designers of BWR plants are summarized in Table 3. The DF values for gross and specific activities, assumed or actually achieved by individual plants, are listed in Table G-1. When electrical conductivity is monitored (see Table G-1), the upper limits are approximately $0.1 \mu\text{mho/cm}$ for the primary coolant and steam condensate and $1 \mu\text{mho/cm}$ for the liquid radioactive waste and fuel pool water.⁷

The DFs for ion exchange systems in PWR plants, especially for the primary coolant system, fluctuate considerably due to the wide variation of pH (4 to 10.5; see Table H-1) and the accompanying variation in the composition. According to the Yankee-Rowe performance data, gross DF for the mixed-bed unit may shift from about 300 at pH 10.5 to 5 at pH 4.5 with the primary coolant containing approximately 900 ppm of boron. The DF for the same unit at pH 7 was in the range of 100 to 200. Table 3 lists DF values that are used by the operators and designers of PWR plants in design calculations, while Table H-1 shows the values of gross DF and DFs for specific nuclides used as a guide by individual plants.

6.0 ORNL LABORATORY-SCALE ION EXCHANGE EXPERIMENTS

Data from ORNL on the performance of a laboratory-scale ion exchange system with radioactive liquids that simulate those at light-water-cooled nuclear power plants are given in Appendix I. These results can be utilized to partially evaluate the performance of mixed-bed and separate-bed ion exchange systems in removing trace radionuclides (^{137}Cs , ^{131}I , and Sr) in the presence of a large excess of nonradioactive chemicals (NaCl , Na_2HPO_4 , H_3BO_3 , and LiOH). Equipment, materials, and experimental conditions used in this work are summarized in Table I-1 of Appendix I. Type 1 and 2 feed solutions simulate BWR and PWR turbine steam condensate containing trace radionuclides and a high concentration of NaCl , which may be present in the condensate side of a leaking seawater-cooled condenser. The composition of type 3 feed solution is in a range between those of the primary and secondary coolants of a PWR. Large amounts of nonradioactive compounds were utilized intentionally to accelerate the breakthrough of ionic species in the feed solution. The radionuclides selected for study were ^{137}Cs , ^{131}I , and ^{90}Sr (for which ^{85}Sr was substituted to simplify radiochemical analysis) since these isotopes are of major concern in nuclear installations. Fresh ion exchange resins were employed in all the experimental runs. These experiments are rather limited in scope in view of the large number and variable amounts of nonradioactive and radioactive nuclides present in actual plant solutions. Nevertheless, the knowledge obtained through these experiments can be used to understand and predict breakthrough behavior.

The main conclusions drawn from these experiments are:

- (1) The breakthrough behavior of trace ionic radionuclides (Cs^+ , I^- , and Sr^{2+}) is subject to the influence of other ionic nuclides (Na^+ , Li^+ , Cl^- , PO_4^{3-}) that are present in high concentrations. It is also affected, to a lesser extent, by the flow rate of the feed. Breakthrough of all the components occurs earlier when either the concentration of the macroconstituent or the feed flow rate is increased.
- (2) The empirical rules of ionic selectivity (Table D-1 in Appendix D) for strong-acid cation and strong-base anion resins of polystyrene matrix are valid with the types of feed solutions used in this study.
- (3) With feed solutions containing large amounts of Na_2HPO_4 , H_3BO_3 , and LiOH and trace radionuclides, the concentrations of the macroconstituents (Li^+ , PO_4^{3-} , and BO_3^{3-}) in the effluent stream can exceed those in the feed, giving DF values less than one.
- (4) The effectiveness of the mixed-bed ion exchange system in removing trace radionuclides from solutions containing large amounts of Na_2HPO_4 , H_3BO_3 , and LiOH is not as high as that of the separate-bed system. Effectiveness is defined here as the volumes of feed processed per volume of cation or anion resin before reaching a specified breakthrough.
- (5) Specific electric conductivities of the solutions employed in this study are closely related to pH values of the solutions. No simple relation between the conductivity or pH and breakthrough curves was found, except that the maximum or the minimum of the conductivity curve may coincide with the peak of a breakthrough curve, e.g., for PO_4^{3-} .

Although these data represent only a limited amount of laboratory experience and not engineering experience in an operating plant, they illustrate an approach by which needed data can be obtained to further explain, clarify, and substantiate operating results.

7.0 SOME OPERATING PROBLEMS WITH ION EXCHANGE SYSTEMS AT NUCLEAR POWER PLANTS

Some problems that are common to ion exchange systems in all industries and some that are unique to the nuclear power industry are discussed briefly in the sections below.

7.1 Chemical and Physical Properties of Ion Exchange Resins in Service

Strong-acid and strong-base resins as listed in Table 2 (of both regular grade and nuclear grade) with approximately 8 to 12% divinylbenzene, a cross-linking agent, are the most widely used types in nuclear power plants. Results obtained by some plants indicate that nuclear-grade resins are subject to breakage and attrition. As both regular and nuclear grades have the same basic structure, it is possible that such breakage is caused by much longer service life, radiation damage, and higher temperature in the nuclear than in the nonnuclear service, rather than being attributable to the difference in resin grade.

The stability of the strong-base anion resins is inferior to that of the strong-acid cation resins. This is especially true of the OH^- -form strong-base resins. At temperatures above 140°F , the amine groups of type 1 anion resins (see Table 2) tend to be leached out by aqueous solution. The temperature limit of type 2 anion resins is even lower than that of type 1 ($\sim 100^\circ\text{F}$). Strong-base resins of the Cl^- and BO_3^{3-} forms are appreciably more stable than those with the OH^- form. For this reason, shipment of such resins in the OH^- form is not commonly practiced.

In PWR plants, the mixed-bed ion exchange system with $\text{H}^+ - \text{OH}^-$ forms is less frequently used in the primary system. This is because the pH of the effluent from this system becomes quite acid at elevated temperatures, thus increasing the rate of corrosion. To reduce corrosion problems, mixed resins of the $\text{NH}_4^+ - \text{OH}^-$ forms were utilized to maintain the pH in the range of 8.5 to 9.5 near room temperature. At high temperatures,

however, the pH of $\text{NH}_4^+ - \text{OH}^-$ form resins approaches the acid range. In addition, nitrogen gas produced by the radiolysis of NH_4^+ creates another operating problem. Therefore, the $\text{NH}_4^+ - \text{OH}^-$ form resins are now being replaced by the $\text{Li}^+ - \text{OH}^-$ form resins; the latter serve to maintain the effluent pH at about 9 to 10.5 near room temperature.

The operating capacity of various ion exchange resins under conditions encountered in nuclear plants may be considered to vary from 50 to 75% of the theoretical capacity, depending upon the regeneration level and other factors. In addition, reduction in the operating capacity results from degradation of the resins by dissolved oxygen and H_2O_2 . Fouling of resins by trapped solid particles and trace organics can not only reduce their capacity but also cause difficulty in regenerating the resins.

7.2 Behavior of Nonionic Nuclides

Nonionic nuclides refer to those that are nonelectrolytes, in colloidal particulate form (10^{-3} - to $1\text{-}\mu$ sizes), or in large ($>1\text{-}\mu$) suspended solid particulate form. Nonionic nuclides may be produced from ionic species through chemical reactions, such as oxidation-reduction. For example, $^{131}\text{I}^-$ ions in a solution containing an appreciable amount of oxygen were not removed by anion exchange resins,²¹ presumably due to the conversion of I^- ions to elemental iodine by oxidation. Because of the nature of the ion exchangers, nonionic nuclides are normally not removable by the ion exchange process and become a source of leakage from the ion exchange system. However, a fraction of the large solid particles is removed by the system with ion exchange resins functioning as a filter. It has been reported²² that certain types of ion exchange resins (e.g., macroreticular type, especially strong-base anion resins) are quite effective for removing colloidal particles (e.g., silica, metallic oxides, etc.) because of their characteristic pore structure. The macroreticular resins are also more resistant to breakage and attrition, but have lower capacities than the gel type resins.

According to a recent study²³ at the Yankee-Rowe plant, solid particles collected from samples of the primary coolant on a series of filters (ranging from 0.02μ to 8μ in pore size) consisted mainly of sizes larger than 5μ ($\sim 78\%$ by weight). Nuclides identified in this crud included ^{51}Cr , ^{181}Hf , ^{54}Mn , ^{59}Fe , ^{58}Co , ^{60}Co , and $^{110\text{m}}\text{Ag}$. Major radioactivities came from ^{59}Fe and ^{58}Co . High specific activities ($\text{dis min}^{-1} \text{mg}^{-1}$) seemed to be associated with particles larger than 5μ and smaller than 0.5μ . Silver- $^{110\text{m}}\text{Ag}$ particles which originated from the control rod material were colloidal in nature since they were in the range from 0.22 to 0.45μ , but they tended to plate out on metallic surfaces.

Examples of nuclides that have a tendency to form colloidal particles are silica, yttrium, rare earths, and molybdenum, as well as metallic oxides of iron, copper, cobalt, and other major corrosion products. The behavior of these nuclides in aqueous solution is quite complex, and an alkaline medium causes hydrolysis which may lead to precipitation, colloid formation, adsorption on oxides, etc. Molybdenum may also form an anion in such an environment.

7.3 Daughters of Short-Lived Radionuclides

Sometimes nuclides appearing in the effluent from an ion exchange system may have short-lived precursors. Consequently, a distorted conclusion may be reached in the evaluation of the performance of the system by merely comparing the effluent and feed concentrations. For example, some Ce, Ba, and La isotopes are the decay products of short-lived noble gas (Xe) precursors. As gaseous nuclides are not removed by ion exchange, decay products formed from gaseous precursors after leaving the ion exchange system may result in higher effluent concentrations of these nuclides than those in the feed, thus giving fractional values of instantaneous DF. On the other hand, some nuclides that are primarily nonionic in character may have ionic precursors. Two good examples are Y and Zr-Nb isotopes, which have Sr isotopes as precursors. The fraction of Y and Zr-Nb produced from decay of strontium isotopes

retained by the ion exchange resins is most likely to remain within the resin particles because of the higher selectivity of the resins for Y and Zr-Nb over divalent Sr.

7.4 Recycling of Effluent from Ion Exchange System

It has been suggested that ion exchange effluent not conforming to the desired or required water quality specifications may be recycled through the same system so as to achieve or exceed the specifications. Breakthrough of a particular species of interest may result from a loaded resin or partially loaded (caused by use or incomplete regeneration) resin. Recycling through the same resin will have little effect on further removal of that particular species. Breakthrough is directly proportional to the concentration of the species on the resin at the bottom of the column or the concentration of the species in the column if the resin is mixed due to back flushing. Breakthrough may also result due to a low distribution coefficient [Eq. (D-16) in Appendix D]. Recycle through the same resin will show little improvement. Breakthrough may also occur because of nonexchangeable species (nonionized nuclides, insoluble particulates, and colloids) which are incapable of undergoing ion exchange. Recycle through the same system will have little effect. A different resin or process must be used, which will interact with the existing species. Even on a fresh resin bed it may be possible to remove 99% of the species of interest; however, the remaining 1% will be in a nonexchangeable form, and recycle would have little effect. The DF of the recycle run diminishes as the interfacial nuclide concentrations between the main liquid stream and the resin particle reached after the first run approach the equilibrium values. Breakthrough may also be caused by exceedingly high or low flow rate of feed. The preferred method of correction is to adjust the flow rate to a proper value so that the desired specifications are obtained the first time, rather than recycling the effluent. In the event this approach should fail to produce an effluent of required quality, little improvement can be expected from recycling.

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APPENDIX A. COMPOSITIONS OF LIQUID RADIOACTIVE SAMPLES

Table A-1. Radiochemical Composition of Primary Coolant in Boiling-Water Reactor Plants

Radionuclide	Probable Source ^a	Probable Form ^b	Average Concentration ($\mu\text{Ci}/\text{ml}$) ^c	
			Dresden ^d	Big Rock Point ^e
³ H	(1), (2)	Water, gas	1.5×10^{-3}	
²⁴ Na	(1)	Cation	2.3×10^{-3}	5.0×10^{-3} (cation)
⁵¹ Cr	(1)	Anion	5×10^{-4}	2.9×10^{-5} (anion)
⁵⁴ Mn	(1)	Cation, s	1.6×10^{-6}	1.1×10^{-5} (45% cation, 55% s)
⁵⁵ Fe	(1)	Cation, s	7.0×10^{-5}	
⁵⁹ Fe	(1)	Cation, s		1.8×10^{-5} (30% cation, 70% s)
⁵⁷ Co	(1)	Cation, s	5.8×10^{-6}	
⁵⁸ Co	(1)	Cation, s	7.8×10^{-3}	2.1×10^{-5} (s)
⁶⁰ Co	(1)	Cation, s	1.2×10^{-3}	3.2×10^{-5} (46% cation, 54% s)
⁶⁴ Cu	(1)	Cation, anion	6.1×10^{-3}	5.1×10^{-3} (cation)
⁶⁵ Zn	(1)	Cation, s	4.0×10^{-6}	1.9×10^{-4} (68% cation, 32% s)
⁸⁹ Sr	(2)	Cation	8.2×10^{-5}	
⁹⁰ Sr	(2)	Cation	1.2×10^{-5}	
⁹¹ Sr	(2)	Cation	1.8×10^{-2}	
⁹² Sr	(2)	Cation		2.4×10^{-2} (cation)
⁹³ Y	(2)	s	5×10^{-3}	
⁹⁵ Zr	(1), (2)	s	4.6×10^{-4}	3.5×10^{-5} (s)
⁹⁷ Zr	(1), (2)	s	5×10^{-3}	
⁹⁵ Nb	(1), (2)	s	2.6×10^{-4}	
⁹⁹ Mo	(1), (2)	Anion	1.4×10^{-3}	3.3×10^{-3} (anion)
^{99m} Tc	(1), (2)	Anion, s	6.7×10^{-2}	2.1×10^{-2} (96% anion, 4% s)
¹⁰³ Ru	(2)	s	2.8×10^{-4}	5.3×10^{-6} (s)
¹⁰⁶ Ru	(2)	s	1.3×10^{-6}	
¹⁰⁵ Rh	(2)	s	2×10^{-3}	
¹²⁴ Sb	(1)	s		5.7×10^{-6} (s)
¹³² Te	(2)	Anion, s	4.1×10^{-4}	
¹³¹ I	(2)	Anion	5.0×10^{-3}	7.6×10^{-3} (anion)
¹³³ I	(2)	Anion	4.4×10^{-2}	
¹³⁵ I	(2)	Anion	8.3×10^{-2}	
¹³⁴ Cs	(2)	Cation	1.8×10^{-5}	1.0×10^{-4} (cation)
¹³⁶ Cs	(2)	Cation	2.7×10^{-5}	
¹³⁷ Cs	(2)	Cation	3.7×10^{-5}	1.6×10^{-4} (cation)
¹⁴⁰ Ba	(2)	Cation	1.2×10^{-3}	3.7×10^{-4} (cation)
¹⁴¹ Ce	(2)	Cation, s	4.7×10^{-4}	3.6×10^{-5} (70% cation, 30% s)
¹⁴³ Ce	(2)	Cation, s	1.6×10^{-3}	
¹⁴⁴ Ce	(2)	Cation, s	1.5×10^{-4}	
¹⁴⁷ Nd	(2)	Cation, s	1.6×10^{-4}	
^{110m} Ag	(1)	s	1.9×10^{-6}	4.9×10^{-6} (s)
¹⁸² Ta	(1)	s	1.0×10^{-5}	
²³⁹ Np	(2)	Cation, s	1.2×10^{-2}	3.4×10^{-4} (cation)
²⁴² Cm	(2)	Cation, s	1.7×10^{-7}	

^a(1) Products of neutron activation of nuclides from fuel cladding, construction materials, and water.
 (2) Leakage from fuel. Mostly fission products.

^bs: insoluble solids.

^cNumbers in parentheses indicate approximate distributions of individual nuclides in different chemical and physical forms.

^dCalculated from data in ref. 4. Dresden 1 has been in operation for about 9 years when these analyses were made. The fuel was slightly enriched UO_2 clad in Zircaloy-2.

^eAdapted from ref. 5. Big Rock Point had been in operation for about 8 years. Fuel cladding material was Zircaloy-2. s: insoluble solids.

Table A-2. Radiochemical Composition of Primary Coolant in Relatively New BWR Plants^{a,b,c}

Radionuclide	Average Concentration ($\mu\text{Ci/ml}$)				
	Oyster Creek	Millstone	Monticello	Dresden 2	Dresden 3
^3H	2.5×10^{-3}		7.7×10^{-3}	3.4×10^{-3}	3.3×10^{-3}
^{14}C	$<1 \times 10^{-6}$				
^{24}Na	1.41×10^{-3}	2.61×10^{-3}	8.4×10^{-4}	8.40×10^{-3}	2.36×10^{-2}
^{32}P	2.9×10^{-4}			1.7×10^{-4}	
^{45}Ca				$<4 \times 10^{-7}$	
^{51}Cr	1.87×10^{-3}	5.47×10^{-3}	2.96×10^{-3}	7.0×10^{-3}	9.79×10^{-3}
^{54}Mn	4.3×10^{-5}	4.51×10^{-5}	1.70×10^{-5}	1.5×10^{-4}	5.90×10^{-5}
^{55}Fe	2.2×10^{-3}			1.11×10^{-4}	
^{59}Fe	2.4×10^{-5}	1.34×10^{-5}	ND ^d	5.02×10^{-5}	4.16×10^{-5}
^{58}Co	3.32×10^{-5}	1.54×10^{-4}	1.91×10^{-4}	1.8×10^{-4}	2.43×10^{-4}
^{60}Co	7.80×10^{-5}	1.37×10^{-4}	1.44×10^{-4}	1.4×10^{-3}	3.37×10^{-4}
^{63}Ni	$<5 \times 10^{-7}$			2.4×10^{-6}	
^{65}Zn	$<1 \times 10^{-5}$	3.18×10^{-5}	7.72×10^{-4}	2.35×10^{-5}	2.05×10^{-5}
^{83}Br		4.63×10^{-4}		1.4×10^{-3}	3.11×10^{-3}
^{84}Br	4.34×10^{-3}	4.95×10^{-3}	1.96×10^{-2}	3.12×10^{-3}	6.27×10^{-3}
^{89}Rb	6.70×10^{-3}	3.70×10^{-3}		2.96×10^{-3}	3.55×10^{-3}
^{89}Sr	1.4×10^{-4}		3.3×10^{-5}	1.21×10^{-5}	
^{90}Sr	5.6×10^{-6}		6.1×10^{-6}	1.6×10^{-7}	
^{91}Sr	3.51×10^{-3}	3.75×10^{-3}	6.63×10^{-3}	1.93×10^{-3}	6.49×10^{-4}
^{92}Sr	2.33×10^{-3}	1.51×10^{-2}	1.44×10^{-2}	6.35×10^{-3}	2.02×10^{-3}
^{91}Y	6.0×10^{-5}			2.4×10^{-5}	
^{92}Y	1.45×10^{-3}		1.32×10^{-2}	2.41×10^{-3}	3.49×10^{-3}
^{93}Y	1×10^{-3}	2.61×10^{-3}	8.12×10^{-3}		
^{95}Zr	$<2 \times 10^{-6}$	3.2×10^{-6}	ND	9.17×10^{-6}	2.23×10^{-5}
^{98}Nb	$<5 \times 10^{-4}$	6.15×10^{-3}	9.3×10^{-3}	ND	1.04×10^{-3}
^{99}Mo	2.1×10^{-3}	2.26×10^{-3}	4.33×10^{-2}	1.40×10^{-3}	2.32×10^{-3}
$^{99\text{m}}\text{Tc}$		1.95×10^{-2}		3.90×10^{-3}	2.43×10^{-2}
^{101}Tc	7.6×10^{-3}	8.77×10^{-2}	1.51×10^{-1}	1.06×10^{-1}	1.09×10^{-1}
^{104}Tc		6.65×10^{-2}	1.11×10^{-1}	5.87×10^{-2}	6.69×10^{-2}
^{105}Ru		1.92×10^{-3}		1.28×10^{-3}	2.32×10^{-3}
^{131}I	3.90×10^{-3}	5.77×10^{-3}	3.45×10^{-2}	6.20×10^{-4}	4.14×10^{-3}
^{132}I	3.28×10^{-2}	2.96×10^{-2}	4.42×10^{-1}	9.20×10^{-3}	4.40×10^{-2}
^{133}I	1.70×10^{-2}	1.33×10^{-2}	1.30×10^{-1}	5.00×10^{-3}	1.80×10^{-2}
^{134}I	8.87×10^{-2}	6.24×10^{-2}	3.59×10^{-1}	4.26×10^{-2}	6.70×10^{-2}
^{135}I	2.34×10^{-2}	1.67×10^{-2}	1.68×10^{-1}	8.84×10^{-3}	1.79×10^{-2}
^{134}Cs	3.33×10^{-5}	3.11×10^{-5}	$<1 \times 10^{-5}$	1.55×10^{-5}	3.70×10^{-5}
^{136}Cs	$<1 \times 10^{-5}$	3.83×10^{-5}	ND	1.50×10^{-5}	3.48×10^{-5}
^{137}Cs	5.66×10^{-5}	6.07×10^{-5}	7.6×10^{-5}	5.31×10^{-5}	3.40×10^{-5}
^{138}Cs	1.50×10^{-2}	1.10×10^{-2}	7.4×10^{-3}		3.92×10^{-5}
^{139}Ba	1.76×10^{-2}	1.16×10^{-2}	1.46×10^{-2}	4.00×10^{-3}	2.53×10^{-3}
^{140}Ba	5.86×10^{-4}	3.48×10^{-4}	7.0×10^{-4}	1.00×10^{-4}	1.33×10^{-4}
^{141}Ba	1.93×10^{-2}	1.68×10^{-2}	9.5×10^{-3}	5.51×10^{-3}	3.50×10^{-3}
^{142}Ba	4.92×10^{-3}	6.56×10^{-3}	ND	1.45×10^{-3}	ND
^{142}La	4.68×10^{-3}	3.50×10^{-3}	5.36×10^{-3}	8.00×10^{-4}	
^{141}Ce	6×10^{-5}	1.98×10^{-5}	$<1 \times 10^{-5}$	ND	2.07×10^{-5}
^{239}Np	1.8×10^{-3}	5.84×10^{-3}	1.32×10^{-2}	1.79×10^{-3}	8.34×10^{-3}
$^{129\text{m}}\text{Te}$	$<1.6 \times 10^{-5}$	ND	ND	ND	ND
$^{131\text{m}}\text{Te}$	$<2.7 \times 10^{-4}$	ND	ND	ND	ND
^{132}Te	$<1.1 \times 10^{-5}$	ND	ND	ND	ND
Gross α	$<1 \times 10^{-7}$	ND	ND	ND	ND

^aAdapted from ref. 6.^bZircaloy-2 was the fuel cladding material used in all the plants listed.^cPlant Name Date of Initial Commercial Operation

Oyster Creek	April 3, 1969
Millstone Point	October 17, 1970
Monticello	March 5, 1971
Dresden 2	March 29, 1970
Dresden 3	March 18, 1971

^dND: not detected.

Table A-3. Radiochemical Composition of Low Purity Liquid Wastes at BWR Plants

Radionuclide	Average Concentration ($\mu\text{Ci}/\text{mL}$)	
	Dresden 1 ^a	Representative BWR ^b
^3H	8.5×10^{-4}	
^{24}Na		2.0×10^{-5}
^{54}Mn	4×10^{-6} (~100%)	2.8×10^{-5}
^{55}Fe	4.8×10^{-5} (~100%)	
^{58}Co	4.5×10^{-4} (49%)	5.6×10^{-5}
^{60}Co	6.0×10^{-4} (77%)	3.5×10^{-5} ; 1.1×10^{-5} ; 1.7×10^{-4}
^{89}Sr	1.5×10^{-4} (20%)	
^{90}Sr	1.6×10^{-5} (6%)	5.6×10^{-7}
^{91}Sr		3.5×10^{-5}
^{92}Sr		4.7×10^{-6}
^{90}Y	1.6×10^{-5} (~100%)	
^{91}Y	1.9×10^{-5} (~100%)	
^{99}Mo		7.0×10^{-5}
$^{99\text{m}}\text{Tc}$		7.2×10^{-5}
^{131}I	2.4×10^{-5} (8%)	7.9×10^{-5} ; 3.8×10^{-5}
^{133}I		5.9×10^{-5}
^{135}I		1.1×10^{-4}
^{134}Cs	3.9×10^{-5} (~0%)	
^{137}Cs	9.9×10^{-5} (~0%)	2.4×10^{-5}
^{140}Ba	9.2×10^{-5} (56%)	1.5×10^{-5}
^{140}La	1.1×10^{-4} (~100%)	1.6×10^{-5}
^{144}Ce	4×10^{-6} (~100%)	

^aAdapted from ref. 4 (Dresden 1). Numbers in parentheses represent amounts of "insolubles" retained by the filter.

^bAdapted from ref. 7.

Table A-4. Radiochemical Compositions of Selected Samples from Fuel Pool Water and Laundry Wastes at the Dresden 1 (BWR) Nuclear Power Plant

Radionuclide	Average Concentration ($\mu\text{Ci/ml}$)	
	Fuel Pool Water ^a	Laundry Wastes ^b
^3H	8.8×10^{-4}	1.3×10^{-6} ; 1.5×10^{-6}
^{54}Mn	8×10^{-6}	
^{58}Co	1.8×10^{-5}	2.2×10^{-6} ; 1.2×10^{-6}
^{60}Co	7.4×10^{-5}	2.3×10^{-6} 6×10^{-7}
^{89}Sr	2×10^{-6}	$<1 \times 10^{-7}$; 9×10^{-7}
^{90}Sr	1.8×10^{-5}	1×10^{-7} ; $<1 \times 10^{-7}$
^{131}I	$<1 \times 10^{-6}$	$<1 \times 10^{-7}$; $<1 \times 10^{-7}$
^{134}Cs	3.1×10^{-4}	6×10^{-7} ; $<1 \times 10^{-7}$
^{137}Cs	7.8×10^{-4}	3.4×10^{-6} ; 8×10^{-7}
^{140}Ba	1.5×10^{-5}	$<1 \times 10^{-7}$; $<1 \times 10^{-7}$
^{141}Ce		$<1 \times 10^{-7}$; $<1 \times 10^{-7}$

^aAdapted from ref. 4.

^bAdapted from ref. 4. For two samples taken on two different dates about two months apart.

Table A-5. Radiochemical Composition of Primary Coolant in Pressurized-Water Reactor Plants

Radionuclide	Probable Source ^a	Probable Form ^b	Average Concentration ($\mu\text{Ci}/\text{ml}$)		
			Yankee-Rowe ^c	Plant X ^d	Plant Z ^e
^3H	(1),(2)	Water, gas	2.4		1.2×10^{-1}
^{14}C			1.2×10^{-5}		
^{24}Na	(1)	Cation	1.9×10^{-2}		
^{32}P			3.3×10^{-5}		
^{35}S			3×10^{-6}		
^{51}Cr	(1)	Anion	3.7×10^{-4}	9.3×10^{-4}	5.3×10^{-3}
^{54}Mn	(1)	Cation, s	2.7×10^{-4}	7.7×10^{-4}	3.8×10^{-5}
^{55}Fe	(1)	Cation, s	1.9×10^{-4}		
^{59}Fe	(1)	Cation, s	1.0×10^{-5}	1.0×10^{-3}	3.0×10^{-5}
^{57}Co	(1)	Cation, s	1.2×10^{-6}		
^{58}Co	(1)	Cation, s	4.7×10^{-4}	2.5×10^{-3}	6.5×10^{-3}
^{60}Co	(1)	Cation, s	7.7×10^{-5}	7.4×10^{-4}	7.3×10^{-4}
^{63}Ni	(1)	Cation, s	8.0×10^{-6}		
^{64}Cu	(1)	Cation, anion, s	5.4×10^{-4}		
^{89}Sr	(2)	Cation	2.8×10^{-6}	3.7×10^{-3}	3.6×10^{-3}
^{90}Sr	(2)	Cation	4×10^{-7}	1.3×10^{-1}	1.8×10^{-4}
^{91}Sr	(2)	Cation	9.8×10^{-5}	1.9×10^{-3}	1.4×10^{-3}
^{90}Y	(2)	s		1.5×10^{-4}	1.1×10^{-3}
^{91}Y	(2)	s		5.9×10^{-3}	1.2×10^{-1}
^{92}Y	(2)	s		7.0×10^{-4}	
^{95}Zr	(1),(2)	s	1.7×10^{-5}	6.9×10^{-4}	1.3×10^{-6}
^{95}Nb	(1),(2)	s	1.9×10^{-5}	6.7×10^{-4}	
^{99}Mo	(1),(2)	Anion	1.2×10^{-4}	5.2	1.9
^{103}Ru	(2)	s	~ 0		3.2×10^{-3}
^{106}Ru	(2)	s	~ 0		1.6×10^{-3}
^{122}Sb	(1)	s	1.0×10^{-4}		
^{124}Sb	(1)	s	2.0×10^{-5}		
^{132}Te	(2)	Anion, s		2.6×10^{-1}	2.2×10^{-2}
^{131}I	(2)	Anion	4.6×10^{-5}	2.5	2.8
^{132}I	(2)	Anion		8.6×10^{-1}	3.6×10^{-1}
^{133}I	(2)	Anion	6.2×10^{-4}	3.8	2.9
^{135}I	(2)	Anion	9×10^{-4}	2.1	
^{134}Cs	(2)	Cation	4.7×10^{-7}	2.5×10^{-1}	1.1×10^{-1}
^{136}Cs	(2)	Cation	~ 0	1.4×10^{-1}	2.6×10^{-1}
^{137}Cs	(2)	Cation	1.1×10^{-6}	1.3	3.9×10^{-1}
^{140}Ba	(2)	Cation	4.7×10^{-6}	4.1×10^{-3}	4.3×10^{-3}
^{141}Ce	(2)	Anion, s	~ 0		
^{143}Ce	(2)	Anion, s	~ 0		
^{144}Ce	(2)	Anion, s	~ 0	3.2×10^{-4}	2.5×10^{-3}
^{143}Pr	(2)	Anion, s			3.8×10^{-3}
^{110m}Ag	(1)	s	1.2×10^{-5}		
^{181}Hf	(1)	s	6×10^{-6}		

Table A-5. Radiochemical Composition of Primary Coolant in Pressurized-Water Reactor Plants (Cont'd)

Radionuclide	Probable Source ^a	Probable Form ^b	Average Concentration ($\mu\text{Ci}/\text{ml}$)		
			Yankee-Rowe ^c	Plant X ^d	Plant Z ^e
^{182}Ta	(1)	s	4.5×10^{-5}		
^{182}Ta	(1)	s	6.2×10^{-5}		
^{185}Ta	(1)	s	1.2×10^{-5}		
^{187}Ta	(1)	s	3.7×10^{-4}		
$^{85\text{m}}\text{Kr}$	(2)	Gas		2.1	1.8
^{85}Kr	(2)	Gas		1.4×10^{-1}	1.1
^{88}Kr	(2)	Gas		3.7	3.1
^{133}Xe	(2)	Gas	8.9×10^{-5}	84	219
^{135}Xe	(2)	Gas	9×10^{-5}	5.9	8.5

^a(1) Neutron activation products of nuclides from fuel cladding, construction material, and water.

(2) Leakage from fuel. Mostly fission products.

^bGas: presumably as dissolved gas.

s: insoluble solids.

^cYankee-Rowe had been in operation for about 9 years. Calculated from data in ref. 8. The fuel was UO_2 (4.9% ^{235}U) clad in stainless steel.

^dPlant X is based on design calculations assuming 1% fuel failure. From ref. 9.

^ePlant Z is based on design calculations assuming 1% fuel failure and other conditions different from Plant X. From ref. 9.

Table A-6. Radiochemical Composition of Selected Samples from Secondary Coolant and Liquid Waste Holdup Tank at the Yankee-Rowe (PWR) Nuclear Power Plant

Radionuclide	Concentration of Selected Samples ($\mu\text{Ci/ml}$) ^a			
	Secondary Loop		Liquid Wastes	
	Steam Generator Blowdown ^b (Continuous)	Turbine Condensate ^c		
³ H	2.5×10^{-3} ; 1.8×10^{-2}	6.3×10^{-3} ; 1.9×10^{-2}	3.8×10^{-1}	
¹⁴ C	$<2 \times 10^{-7}$; $<1 \times 10^{-7}$	3×10^{-7} ; $<2 \times 10^{-7}$	1.2×10^{-4}	
³² P	$<2 \times 10^{-7}$; $<1 \times 10^{-8}$	$<2 \times 10^{-7}$; $<1 \times 10^{-8}$	$<2 \times 10^{-7}$	
⁵¹ Cr	3×10^{-6} ; $<1 \times 10^{-7}$	$<5 \times 10^{-7}$	$<5 \times 10^{-7}$	
⁵⁴ Mn	1×10^{-6} ; 1×10^{-7}	1.2×10^{-6} ; $<1 \times 10^{-7}$	1.4×10^{-3}	
⁵⁵ Fe	5×10^{-8} ; $<5 \times 10^{-8}$	$<5 \times 10^{-8}$; $<5 \times 10^{-8}$	8.2×10^{-4}	
⁵⁹ Fe	1×10^{-6} ; $<1 \times 10^{-7}$	$<5 \times 10^{-8}$; $<1 \times 10^{-7}$	1.0×10^{-6}	
⁵⁷ Co			6×10^{-6}	
⁵⁸ Co	2×10^{-6} ; $<1 \times 10^{-7}$	8×10^{-7} ; $<1 \times 10^{-7}$	2.9×10^{-4}	
⁶⁰ Co	5×10^{-7} ; $<5 \times 10^{-8}$	1×10^{-7} ; $<1 \times 10^{-7}$	6.4×10^{-4}	
⁶³ Ni	2×10^{-7} ; $<1 \times 10^{-8}$	$<1 \times 10^{-7}$; 4×10^{-8}	2.3×10^{-4}	
⁸⁹ Sr	$<1 \times 10^{-8}$; $<5 \times 10^{-8}$	$<2 \times 10^{-8}$; $<2 \times 10^{-8}$	3×10^{-7}	
⁹⁰ Sr	5×10^{-9} ; $<2 \times 10^{-8}$	1×10^{-8} ; $<1 \times 10^{-8}$	7×10^{-8}	
⁹⁵ Zr	8×10^{-7} ; $<1 \times 10^{-7}$	$<1 \times 10^{-7}$		
⁹⁵ Nb	6×10^{-7} ; $<1 \times 10^{-7}$	$<1 \times 10^{-7}$		
^{110m} Ag	2×10^{-7} ; $<2 \times 10^{-8}$	$<2 \times 10^{-7}$; $<1 \times 10^{-7}$	3.1×10^{-4}	
¹²⁴ Sb	1×10^{-7}			
¹³¹ I	$<4 \times 10^{-8}$; 5×10^{-7}	$<5 \times 10^{-7}$; 5×10^{-7}	$<3 \times 10^{-8}$	
¹³⁴ Cs	$<2 \times 10^{-8}$; $<6 \times 10^{-8}$	$<1 \times 10^{-7}$; $<6 \times 10^{-8}$	2.0×10^{-5}	
¹³⁷ Cs	1×10^{-8} ; $<1 \times 10^{-8}$	2×10^{-8} ; $<1 \times 10^{-8}$	5.4×10^{-5}	
¹⁸² Ta			1.6×10^{-5}	

^aAdapted from ref. 8.

^bSamples taken on two different dates approximately 20 months apart.

^cSamples taken on two different dates approximately 11 months apart.

APPENDIX B. ION EXCHANGE SYSTEMS AT THE TEST REACTOR AREA (TRA),
NATIONAL REACTOR TESTING STATION (NRTS), IDAHO

Two test reactors at the Test Reactor Area (TRA) site, the Advanced Test Reactor (ATR) and the Engineering Test Reactor (ETR), are similar to a PWR with regard to design except that the fuel cladding material is aluminum rather than zircaloy, the reactivity is regulated entirely with control rods rather than with chemical shim, and no power generating facilities are involved. The ATR and ETR have ion exchange systems for the purge stream from the primary coolant. Both systems are of the same design except that the size of the ETR system is approximately one-half that for the ATR. An additional difference between the two is the method of pH adjustment. As shown in Fig. B-1, there are two ion exchange units in each system, each unit consisting of one cation and one anion column. Normally, only one unit is used in the ATR, whereas both units are employed continuously in the ETR.

The pH value of the feed to the ATR and the ETR ion exchange systems is controlled at 5 ± 0.2 in each case. There is a tendency for the pH of the primary coolant to increase above this value, and HNO_3 is employed to adjust the pH in the ATR. In the ETR, pH adjustment is carried out by proper routing of the feed through the ion exchange system. That is, in the event the pH should drop below the control point, the feed bypasses the cation exchange unit because the exchange reaction with resins of H^+ form tends to lower the pH of the solution further. The anion exchange unit is bypassed if the pH increases above 5 ± 0.2 .

Sodium-24 is the predominant radionuclide in the primary coolant; the other radionuclides consist of fission products and activated corrosion products. The main source of ^{24}Na is the $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$ reaction, which occurs in the aluminum from the fuel cladding. Table B-1 illustrates the radiochemical composition of selected batches of the primary coolant in both the ATR and the ETR. As is the case with BWRs and PWRs, the sample composition can vary from one sampling period to another. However, compositions of the primary coolant in both the ATR and the ETR are nearly identical, and no long-lived fission products (^{90}Sr , ^{134}Cs , ^{137}Cs) are present either in the ATR or the ETR. Isotopes of such corrosion products as Co and Ni are absent.

PRIMARY COOLANT FLOW:
 NORMAL 46,000 gpm
 CAPACITY 50,000 gpm
 BYPASS CLEAN-UP FLOW:
 ATR { NORMAL 200 gpm (ONE UNIT)
 { MAXIMUM 480 gpm (BOTH UNITS)
 ETR: NORMAL 200 gpm (BOTH UNITS)

FOR ATR:
 ——— NORMAL FLOW
 - - - - FLOW THROUGH
 SPARE UNIT
 FOR MAXIMUM FLOW BOTH
 UNITS ARE USED
 FOR ETR: BOTH UNITS ARE USED
 CONTINUOUSLY

Fig. B-1. Schematic Diagram of the Ion Exchange Systems for the ATR or the ETR Reactor Coolant Bypass Stream.

Table B-1. Radiochemical Composition of Primary Coolant in
Test Reactor Area (TRA) Reactors ($\mu\text{Ci/ml}$)

Radionuclide	Advanced Test Reactor (ETR)	Engineering Test Reactor (ETR)
^{24}Na	2.0×10^{-1}	1.3×10^{-1}
^{51}Cr	6.9×10^{-3}	8.1×10^{-4}
^{56}Mn	3.1×10^{-3}	6.6×10^{-3}
^{64}Cu	5.2×10^{-3}	3.1×10^{-3}
^{91}Sr	3.9×10^{-3}	4.0×10^{-3}
^{92}Sr	7.0×10^{-3}	6.2×10^{-3}
$^{99\text{m}}\text{Tc}$	8.1×10^{-3}	9.2×10^{-3}
^{132}Te	1.2×10^{-4}	3.4×10^{-4}
^{131}I	2.6×10^{-4}	5.4×10^{-4}
^{132}I	2.5×10^{-3}	6.3×10^{-3}
^{133}I	3.3×10^{-3}	7.6×10^{-3}
^{134}I	1.4×10^{-2}	$1.1 \times 10^{-1\text{a}}$
^{135}I	3.4×10^{-3}	8.9×10^{-3}
^{139}Ba	2.5×10^{-2}	2.8×10^{-2}
^{140}Ba	2.8×10^{-4}	6.2×10^{-3}
^{141}Ce	1.0×10^{-4}	1.2×10^{-4}
^{142}La	3.7×10^{-3}	4.2×10^{-3}
^{143}Ce	3.8×10^{-4}	4.6×10^{-4}
^{144}Ce	2.1×10^{-4}	1.2×10^{-4}
^{188}Re	1.5×10^{-3}	5.1×10^{-4}
^{239}Np	4.7×10^{-4}	1.2×10^{-3}

^aNormally much lower.

The performances of both the ATR and the ETR ion exchange systems (see Table B-2 for the resin type and other information) in removal of radionuclides are evaluated periodically by monitoring the following variables at the indicated time intervals:

- (1) Daily composite samples: gross α in primary coolant system (PCS) and purge stream (PS), gross ($\beta+\gamma$) in PS, and ^{24}Na in PCS.
- (2) Six times per day: gross ($\beta+\gamma$) in PCS.
- (3) Twice per week: ^{133}I and ^{135}I contents of PCS.
- (4) Once a month (composite samples): complete γ -scan of PS, ^{89}Sr , ^{90}Sr , and tritium in PS.
- (5) Twice per month: complete γ -scan of PCS.

Other operating parameters monitored include DF for gross radioactivity, pressure drop across the resin bed (ΔP), electrical conductivity of liquid, ion exchange capacity, and pH. The operating ranges of these parameters under normal conditions are summarized in Table B-3. Limits of various parameters that determine when the resins should be replaced or regenerated are also listed in Table B-3. In the actual operation of the cation bed, the ion exchange capacity of the resin is used as the controlling parameter for such a decision. The electrical conductivity and pH are the controlling parameters for the anion bed. Under ordinary circumstances, gross DFs of up to 15 have been achieved by the end of the loading cycle. The data in Table B-4 demonstrate that the cation exchange unit effectively removes a number of nuclides including Na, Mn, Cu, Sr, Ba, La, and Np. On the other hand, Cr, Tc, Te, and I are removed by the anion exchanger unit, indicating that these nuclides are present primarily as anions. The observation that Y (not shown in Table B-4), W, Ce, and Re are not removed by the cation exchanger or by the anion exchanger presumably implies that they are not in ionic form. The reasons why five nuclides were not detectable in the feed stream but analyses showed their presence in effluents might have been due to suppression of the spectra of these nuclides by predominant spectra of other nuclides in the feed, such as ^{24}Na . A typical radiochemical composition of the exhausted ion exchange resins (after four months of

Table B-2. Basic Technical Data for Ion Exchange Systems at the Test Reactor Area (TRA)

	Advanced Test Reactor (ATR)	Engineering Test Reactor (ETR)
Function of ion exchanger	Decontamination of reactor coolant purge stream	
Number of ion exchangers	2 units in parallel/reactor; 1 cation and 1 anion/unit	
Resin type	Cation: Macroreticular strong acid (Na ⁺ form) Anion: Nuclear grade strong base (OH ⁻ form)	
Exchange capacity (meq/ml wet) (fresh resin)	Cation: 1.57 - 1.72 Anion: 0.62 - 0.71	
Regeneration	Cation: No Anion: Yes	
Regenerating agent	Anion: 4 wt % NaOH	
Resin volume per column, ft ³	70 ft ³ /column	30 ft ³ /column
Design pressure, psig	360	
Normal operating flow rate of feed, gpm	200-240 for each unit	200-240 for 2 units
pH	4.2 - 5.6	
Temperature of feed	125°F	

Table B-3. Normal Ranges and Limits of Operating Parameters for the Advanced Test Reactor (ATR) and Engineering Test Reactor (ETR) Ion Exchange Systems

Parameters Being Monitored	Advanced Test Reactor (ATR)				Engineering Test Reactor (ETR)			
	Normal Oper. Range		Discard Limits		Normal Oper. Range		Discard Limits	
	Cation	Anion	Cation	Anion	Cation	Anion	Cation	Anion
DF of gross activity	8-15	3-6	6	3	2-15	3-6	1.6	2.5
ΔP at 200 gpm, psi	4-8	4-8	20	20	5-15	1-6	20	20
Elec. conductivity, μmho	6-9	0.5-2	11	3.5	1-5	0.2-1	None	3.5
Ion exchange capacity, meq/g	4-4.5	2-3.5	0.6	33% of original	4-4.5	2-3.5	0.6	33% of original
pH	4.2-5.5	5.5-8.0	None	>5.2	5.0-5.6	6.4-9.0	None	>6.4

Table B-4. Radiochemical Compositions of Feed and Effluent Streams for the Advanced Test Reactor (ATR)² Ion Exchange System, and Decontamination Factors

Radionuclide	Radioactivity and Decontamination Factor (DF) ^b					Overall DF
	Feed ($\mu\text{Ci/ml}$)	Cation Column Effluent		Anion Column Effluent		
		$\mu\text{Ci/ml}$	DF	$\mu\text{Ci/ml}$	DF	
²⁴ Na	1.98×10^{-1}	n.d. ^c	•	8.59×10^{-3}	-	23.0
⁴¹ A	7.86×10^{-4}	5.67×10^{-3}	0.14	4.84×10^{-3}	1.17	0.16
⁵¹ Cr	6.86×10^{-3}	7.54×10^{-3}	0.91	1.81×10^{-4}	41.6	37.9
⁵⁶ Mn	3.11×10^{-3}	n.d.	•	n.d.	-	•
⁶⁴ Cu	5.19×10^{-3}	n.d.	•	7.81×10^{-4}	-	6.64
^{85m} Kr	1.08×10^{-4}	1.41×10^{-3}	0.08	1.32×10^{-3}	1.07	0.08
⁸⁸ Kr	3.73×10^{-4}	2.94×10^{-3}	0.01	2.86×10^{-3}	1.03	0.13
⁹¹ Sr	3.69×10^{-3}	n.d. ^c	•	2.49×10^{-4}	-	15.6
⁹⁰ Sr	7.00×10^{-3}	n.d. ^c	•	n.d.	-	•
^{99m} Tc	8.10×10^{-3}	8.86×10^{-3}	0.91	2.97×10^{-5}	298	272
¹³² Te	1.24×10^{-4}	5.14×10^{-5}	2.41	n.d.	•	•
¹³¹ I	2.59×10^{-4}	2.68×10^{-4}	0.97	n.d.	•	•
¹³² I	2.51×10^{-3}	2.40×10^{-3}	1.04	n.d.	•	•
¹³³ I	3.27×10^{-3}	2.84×10^{-3}	1.13	n.d.	•	•
¹³⁴ I	1.36×10^{-2}	1.15×10^{-3}	1.18	n.d.	•	•
¹³⁵ I	3.35×10^{-3}	3.65×10^{-3}	0.92	n.d.	•	•
¹³³ Xe	8.11×10^{-5}	4.03×10^{-4}	0.20	7.74×10^{-4}	0.52	0.10
¹³⁵ Xe	1.52×10^{-3}	6.00×10^{-3}	0.25	6.46×10^{-3}	0.93	0.24
¹³⁹ Ba	2.50×10^{-2}	1.37×10^{-3}	18.2	2.38×10^{-3}	0.58	10.5
¹⁴⁰ Ba	2.78×10^{-4}	n.d.	•	2.43×10^{-5}	-	11.4
¹⁴¹ Ce	1.03×10^{-4}	1.16×10^{-4}	0.89	1.11×10^{-4}	1.04	0.93
¹⁴² La	3.70×10^{-3}	n.d.	•	1.42×10^{-3}	-	2.61
¹⁴³ Ce	3.75×10^{-4}	2.86×10^{-4}	1.31	2.84×10^{-4}	1.30	1.32
¹⁴⁴ Ce	2.11×10^{-4}	1.62×10^{-5}	13.0 ^d	2.70×10^{-5}	0.60	7.81
¹⁸⁸ Re	1.48×10^{-3}	1.81×10^{-3}	0.82	3.86×10^{-3}	0.47	0.38
²³⁹ Np	4.68×10^{-4}	n.d.	•	n.d. ^c	-	•
⁴⁶ Sc	n.d. ^c	7.83×10^{-6}	-	7.02×10^{-6}	1.11	-
⁸⁷ Kr	n.d. ^c	4.05×10^{-3}	-	3.14×10^{-3}	1.29	-
¹⁰³ Ru	n.d. ^c	5.40×10^{-6}	-	3.24×10^{-6}	1.67	-
¹⁸⁷ W	n.d. ^c	1.76×10^{-4}	-	1.08×10^{-4}	1.63	-
¹⁴⁷ Np	n.d. ^c	n.d. ^c	-	4.86×10^{-5}	-	-

^aThe composition of this test reactor is not directly related to that of power reactors.

^bDF is defined as the ratio of the feed activity to the effluent activity. It has not been corrected for the reduction in the activity due to radioactive decay. A DF of • simply means that the decontamination is nearly complete since the effluent activity is not detectable. The activity of feed to the anion exchanger is considered to be the same as that of effluent from the cation exchanger.

^cNot detectable.

^dSeveral other samples showed almost negligible removal of Ce by both cation and anion exchangers.

operation) from the ETR appears in Table B-5. Although cobalt and lanthanum are shown in the table, their radiochemical concentrations in the feed stream to the ion exchanger system were normally very low.

Table B-5. Typical Radiochemical Analysis of Exhausted Ion Exchange Resins from the Engineering Test Reactor (ETR)

Nuclide	Radioactivity ($\mu\text{Ci/g}$ wet resin)	
	Cation Resin	Anion Resin
^{58}Co	0.7	-
^{60}Co	0.92	-
^{140}La	0.046	-
^{131}I	-	0.4
^{132}I	-	0.15
^{133}I	-	0.33
^{135}I	-	0.03

APPENDIX C. ION EXCHANGE SYSTEMS AT THE IDAHO CHEMICAL PROCESSING
PLANT (ICPP), NATIONAL REACTOR TESTING STATION (NRTS), IDAHO

- C.1 Ion Exchange System for Waste Evaporator Condensate
- C.2 Ion Exchange System for Fuel Storage Basin

APPENDIX C. ION EXCHANGE SYSTEMS AT THE IDAHO CHEMICAL PROCESSING PLANT (ICPP), NATIONAL REACTOR TESTING STATION (NRTS), IDAHO

The Idaho Chemical Processing Plant is one of four AEC-owned and -administered nuclear fuel reprocessing facilities for the primary purpose of recovering uranium from spent fuel. Its special feature is the capability for handling highly enriched fuel (20 to 93% ^{235}U). Two ion exchange systems are in use at ICPP, one in the waste evaporator area and the other at the fuel storage basin. The former employs an organic type of synthetic resin, whereas an inorganic type of ion exchange material is used in the latter.

C.1 Ion Exchange System for Waste Evaporator Condensate

This system (Fig. C-1) is made up of two cation exchange columns (2 ft diam by 3 ft high, fabricated of stainless steel type 347), each containing 6 ft³ of strong-acid resins of the macroreticular type. The two columns can be operated either in series or in parallel. This type of resin has been shown to possess high physical and chemical stability and to be quite efficient in removing ^{137}Cs , ^{90}Sr , and ^{144}Ce from ICPP evaporator condensate having very low pH values (1.2 to 2.6). It is not very effective in removing ^{106}Ru and $^{95}\text{Zr-Nb}$. Although no specific physical and chemical forms of ^{106}Ru and $^{95}\text{Zr-Nb}$ have been determined, they are present in the condensate, presumably in the form of complex anions. The typical radiochemical composition of the evaporator condensate appears in Table C-1, which indicates ^{137}Cs , ^{90}Sr , and ^{144}Ce to be the major nuclides. This is in sharp contrast with the primary coolant in the ATR and the ETR, where the predominant nuclides are short-lived radioisotopes. Pertinent technical data for the ICPP evaporator condensate ion exchange system are given in Table C-2.

Strontium-90 and cesium-137 are among the major radionuclides present in the waste evaporator condensate, as shown in Table C-1. Strong-acid cation resins have a very high selectivity for ^{90}Sr over ^{137}Cs in the condensate. Thus, no breakthrough of ^{90}Sr has been detectable until a feed throughput of 3500 bed volumes is reached, as shown in Table C-3. No breakthrough of ^{144}Ce was observed through 6000 bed volumes of feed.

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Fig. C-1. ICPP Condensate Decontamination System.

Table C-1. Radiochemical Composition of Idaho Chemical Processing Plant (ICPP) Evaporator Condensate

Radionuclide	No. of Samples	Concentration ($\mu\text{Ci/ml} \times 10^4$)		
		Average	Minimum	Maximum
Gross β	32	11.00	1.20	48.00
^{137}Cs	30	1.50	0.23	4.50
^{90}Sr	32	1.40	0.26	8.90
^{106}Ru	28	0.68	0.17	1.70
^{144}Ce	26	2.00	0.14	13.00
^{95}Zr - ^{95}Nb	18	0.42	0.07	0.75
^{125}Sb	3	<0.28		
pH	32	2.08	1.20	2.61

Table C-2. Basic Technical Data for Ion Exchange System
at the Idaho Chemical Processing Plant (ICPP)

	Evaporator Condensate Ion Exchange System	Fuel Basin Ion Exchange System
Function of ion exchanger	Decontamination of waste evaporator condensate	Decontamination of fuel- storage basin water
Number of ion exchangers	2 (in series)	1
Resin type	Strong-acid cation (Na^+ form)	Zeolite-type inorganic exchanger
Exchange capacity (meq/ml wet) (fresh resin)	1.75	2.26 meq/g dry
Regeneration	Yes	Yes
Regenerating agent	40% NaNO_3	$(\text{NH}_4)_2\text{SO}_4$
Resin volume per column, ft^3	6 each	4
Design pressure, psig	200	
Normal operating flow rate of feed, gpm	9.22 (3 gpm/ ft^2)	14-18
pH	1.2-2.6	7.5-8.5
Temperature of feed	25-60°C	25°C

Table C-3. Performance of the ICPP Ion Exchange System (Strong-Acid Cation Resin, Na⁺ Form) for Evaporator Condensate^a

No. of Bed Vol. of feed ^b	Breakthrough (%)			Decontamination Factor (DF)		
	Total β	¹³⁷ Cs	⁹⁰ Sr	Total β	¹³⁷ Cs	⁹⁰ Sr
1000	15	Background	Background	6.7	-	-
2500	15	2	Background	6.7	50	-
3000	17	3	Background	5.9	33	-
3500	18	6	0.04	5.6	17	2.5 x 10 ³
4000	21	13	0.06	4.8	7.7	1.7 x 10 ³
5000	60	70	0.15	1.7	1.4	6.7 x 10 ²

^apH of ~1 to 2 as interpolated from the plant data.

^bCumulative throughput of feed as multiples of the ion exchange bed volume.

Breakthrough of the total beta activity in the early stage (up to ~3000 bed volumes) of the process appears to be attributable to $^{95}\text{Zr-Nb}$, ^{106}Ru , and other beta emitters. Since ^{106}Ru and $^{95}\text{Zr-Nb}$ are removed primarily by other techniques, the need for regeneration of ion exchange resins is determined by the breakthrough data of ^{137}Cs . Regeneration of resins is carried out using 40% NaNO_3 , normally after approximately 3000 bed volumes (~135,000 gal) of feed throughput.

Distribution coefficients of both ^{137}Cs and ^{90}Sr are favored by an increase in the pH up to 4 (Table C-4). The improvement in the distribution coefficient is especially remarkable for ^{90}Sr , involving a 70-fold change (from 6×10^3 to 4×10^5) when the pH increases from 1 to 4. In contrast, the same change in the pH merely doubled the distribution coefficient for ^{137}Cs from 1.3×10^3 to 2.6×10^3 .

C.2 Ion Exchange System for Fuel Storage Basin

This system consists of one 4-ft³ column (2 ft diam x 5 ft high) containing Zeolon-900 inorganic ion exchanger. The radioactive contaminants in the fuel basin arise primarily from the cutting of various fuel rods and have a typical gross beta activity of approximately 0.108 $\mu\text{Ci/ml}$, the major nuclides being ^{137}Cs ($4.05 \times 10^{-2} \mu\text{Ci/ml}$) and ^{90}Sr ($2.97 \times 10^{-2} \mu\text{Ci/ml}$). The normal feed flow rate through the system ranges from 14 to 18 gpm. The feed passes through a filter (fiber-glass bags covered with nylon sacks and coated with diatomaceous earth) before entering the ion exchange system. The concentrations of ^{137}Cs and ^{90}Sr in the ICPP fuel storage basin cited above are approximately 50 and 1000 times higher, respectively, than those in the BWR fuel storage pool shown in Table A-4 of Appendix A. Approximately 40,000 bed volumes of feed can be treated before 50% breakthrough of ^{137}Cs . At this point, the ion exchanger is regenerated with ammonium sulfate solution. A program is under way to improve the ion exchange system by adding a unit containing clinoptilolite that removes both ^{137}Cs and ^{90}Sr quite effectively. The feed solution will first enter the zeolite unit, the primary function of which is removal of ^{137}Cs . Major removal of ^{90}Sr will be accomplished by feeding the effluent from the zeolite unit through the clinoptilolite unit.

Table C-4. Distribution Coefficients of ^{137}Cs and ^{90}Sr as a Function of Strong-Acid Cation Resin, H^+ Form pH

pH ^a	Distribution Coefficient ^b	
	^{137}Cs	^{90}Sr
1	1.3×10^3	6×10^3
3.04	2.1×10^3	1×10^5
4.04	2.6×10^3	4×10^5

^aAdjusted by addition of NaOH.

^bEstimated from experimental data. See Eq. (D-16) in Appendix D for the definition.

APPENDIX D. FUNDAMENTALS OF ION EXCHANGE

- D.1 Structure of Ion Exchange Particles
- D.2 Ion Exchange Reactions
- D.3 Important Properties of Ion Exchange Resins
 - D.3.1 Exchange Capacity
 - D.3.2 Degree of Cross-Linking
 - D.3.3. Swelling Equilibrium
 - D.3.4 Ionic Selectivity
 - D.3.5 Factors Affecting Ionic Selectivity
 - D.3.6 Stability
- D.4 Kinetics and Mechanism of Ion Exchange Process
- D.5 Leakage in Ion Exchange System
 - D.5.1 Incomplete Regeneration
 - D.5.2 Exceedingly High or Low Feed Flow
 - D.5.3 Organic Fouling of Ion Exchange Bed

APPENDIX D. FUNDAMENTALS OF ION EXCHANGE

Ion exchange refers to a process in which a reversible interchange of ions of the same sign takes place in a stoichiometrically equivalent proportion between an electrolyte solution and a solid phase (ion exchanger) in contact with the solution. No permanent alteration in the structure of the solid phase is involved in such an exchange. For example, with a cation exchanger:

while with an anion exchanger:

Here, $\bar{R}^-$ and $\bar{R}^+$ represent the basic structures of the cation and anion exchangers, respectively, containing single exchange sites. The exchangeable positive and negative ions, X^+ and Z^- , respectively, are frequently called counterions.

D.1 Structure of Ion Exchange Particles

The physical structure of ion exchange resin particles may be visualized by the schematic diagram of a single particle of hydrated strong-acid cation resin depicted in Fig. D-1. The particle consists of interwoven chains of styrene polymers with divinylbenzene (DVB) cross-linking the chains, the void space being filled with water. Fixed exchange sites [equivalent to $\bar{R}^-$ in Eq. (D-1); e.g., $-SO_3^-$] are represented by $\ominus$, and exchangeable cation sites [equivalent to X^+ in Eq. (D-1); e.g., H^+ , Li^+ , NH_4^+ , etc.] are represented by $\oplus$. A similar sketch is applicable to a single hydrated anion exchange resin particle, except that fixed exchange

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Fig. D-1. Schematic Diagram of Strong-Acid Cation Exchange Resin (Adapted from ref. 10).

sites are now positively charged (e.g., quaternary ammonium radical, $-\text{CH}_2\text{NR}_3^+$) while exchangeable sites are negatively charged (e.g., Cl^- , OH^- , etc.)

D.2 Ion Exchange Reactions

Equations (D-1) and (D-2) demonstrate that ion exchange reactions between ionic species in an electrolyte solution and counterions in the resin generally take place reversibly. The forward reaction occurs, for example, when fresh cation and anion exchange resins are brought in contact with waste solution containing Cs^+ and I^- ions. This step is usually called the exhaustion (or loading) phase of the ion exchange process. Additional examples of reactions in the exhaustion phase are shown below:

For cation resins,

and

For anion resins,

and

The reverse reactions take place as exhausted resins (having most of the exchangeable counterions in the resins replaced) are treated with a concentrated solution of exchangeable counterions. This is the regeneration phase of the ion exchange process. Typical regenerant solutions are dilute H_2SO_4 , NaOH , and NH_4OH . Such reactions are illustrated below:

For cation resins,

and

For anion resins,

and

D.3 Important Properties of Ion Exchange Resins

D.3.1 Exchange Capacity

Exchange capacity refers to the total number of sites available for exchange in the ion exchanger and is commonly expressed as milliequivalents (meq) per milliliter of the wet resin. Total (or theoretical) capacity is the maximum available capacity of the resin, while operating capacity

represents the actual useful capacity obtainable under given operating conditions. The units normally used to express the operating capacity in the nonradioactive water treatment are kilograins of CaCO_3 per cubic foot of resin ($\text{kgrains CaCO}_3/\text{ft}^3$).

When applied to a column operation, the operating capacity is also termed the breakthrough capacity which specifies the capability of an ion exchange column to remove the component or components of interest from a feed solution to a desired concentration level in the effluent. The operating (or breakthrough) capacity of a given type of ion exchange resin is determined by numerous factors, among which are: (1) the regeneration level (the extent to which the resin has been converted to the desired ionic form); (2) the flow rate, pH, and temperature of the feed solution; (3) the composition of the feed solution; (4) the selectivity of the resin for one ion with respect to another; (5) the resin particle size; and (6) the extent of cross-linkage in the basic resin structure. The ways in which factors 1 and 6 influence the exchange capacity are illustrated in Figs. D-2 and D-3.

The degree of column utilization, defined as the ratio of the breakthrough capacity to the total capacity, is often employed to express the efficiency with which the ion exchange column has been used. The steeper the breakthrough curve (a plot of time or the cumulative effluent volume vs the ratio of the concentration of a component in the effluent, C_e , to the concentration of the same component in the feed, C_f), the higher is the degree of column utilization. This is illustrated qualitatively in Fig. D-4. The reciprocal of the fractional breakthrough is known as the instantaneous decontamination factor, $(\text{DF})_i$. That is,

$$(\text{DF})_i = C_f / C_e \quad . \quad (\text{D-11})$$

D.3.2 Degree of Cross-Linking

Degree of cross-linking describes the extent to which the linear polymer chains (e.g., polystyrene) are joined by a cross-linking agent (e.g., divinylbenzene) (see Fig. D-1 for a physical concept of cross-linking). However, the degree of cross-linking generally employed in

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Fig. D-2. Effect of Regeneration Level on Operating Capacity (Strong-Acid Cation Resin, H⁺ Form).

Fig. D-3. Effect of Cross-Linkage on Total Capacity (Strong-Acid Cation Resin, H⁺ Form).

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Fig. D-4. Breakthrough Curves (Schematic).

characterizing a type of ion exchange resin is not a directly measured quantity. Rather, it is a qualitative measure based on the nominal weight percent of the cross-linking agent in the initial monomer mixture. For ion exchange resins with the polystyrene matrix, it is normally expressed in terms of the weight percent DVB in polymer resins. The polystyrene-DVB-type ion exchange resins commonly used in nuclear power plants have DVB concentrations from 8 to 12%.

D.3.3 Swelling Equilibrium

When dry ion exchange resin particles are placed in contact with a solvent, they absorb solvent and swell to a limited extent. This solvent-retention capacity is a reproducible equilibrium quantity and depends on properties of the resin such as functional group, ionic form, total capacity, cross-linking, and solvent-polymer interaction.

D.3.4 Ionic Selectivity

Ion exchangers of a given type (cation or anion) in contact with a solution containing two or more species of the same ionic charge generally exhibit a different preference for exchange with each ionic species present during equilibration. As a result, the relative amounts of the various species in the ion exchanger would be different from those in the solution. For example, ion exchange reactions between strong-acid cation resins (H^+ form) and a dilute aqueous solution containing Cs^+ and Na^+ ions may be represented by:

and

Chemical analyses of the resins and the solution will reveal that the Cs^+/Na^+ concentration ratio in the resin phase is greater than that in the solution, implying that the resin has a preference for Cs^+ .

A set of equilibrium data (ion exchange isotherm) depicted in Fig. D-5 may be obtained by repeated equilibrations of the resins with solutions containing different concentrations of ionic species A and B at a known temperature. Here, the equivalent ionic fraction of ion A is defined as follows: For the resin phase,

$$y_A = \frac{n_A y_A}{n_A y_A + n_B y_B} ; \quad (\text{D-14})$$

for the solution phase,

$$x = \frac{n_A x_A}{n_A x_A + n_B x_B} , \quad (\text{D-15})$$

where n is the electrochemical valence, y is the concentration in the resin phase (moles per gram of resin), and x is the corresponding concentration in the solution phase (moles per gram of solvent; moles per milliliter of solution is commonly used in aqueous solution).

The tendency of ion exchangers to prefer certain ionic species over others is normally expressed by one of the quantities defined below:

$$\text{Distribution coefficient (for A)} = K_d = \frac{y_A}{x_A} , \quad (\text{D-16})$$

$$\text{Selectivity coefficient (for A over B)} = K_B^A = \left(\frac{y_A}{x_A} \right)^{|n_B|} / \left(\frac{y_B}{x_B} \right)^{|n_A|} , \text{ and } (\text{D-17})$$

$$\text{Separation factor (for A over B)} = \alpha_B^A = \left(\frac{y_A}{x_A} \right) / \left(\frac{y_B}{x_B} \right) = \left(\frac{y_A}{x_A} \right) / \left(\frac{y_B}{x_B} \right) . \quad (\text{D-18})$$

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Fig. D-5. Representation of Ion Exchange Equilibrium (Ion Exchange Isotherm).

These three quantities are, in general, different from one another. In the literature, however, values of α_B^A are often given as K_B^A without any specific definition. It can be seen from Eqs. (D-17) and (D-18) that $K_B^A = \alpha_B^A$ only in the case dealing with monovalent ions. For a system containing competing ionic species of different valences, α_B^A varies considerably with experimental conditions (e.g., total ionic concentration) while K_B^A is only slightly affected.

The value of α_B^A at a given value of X_A may be evaluated from the ion exchange isotherm (Fig. D-5) by direct measurement of areas $\overline{abcd}$ and $\overline{aefg}$ because from Eq. (D-18) and Fig. D-5:

$$\alpha_B^A = \frac{Y_A X_B}{X_A Y_B} = \frac{Y_A (1 - X_A)}{X_A (1 - Y_A)} = \frac{\text{Area } \overline{abcd}}{\text{Area } \overline{aefg}} \quad (\text{D-19})$$

D.3.5 Factors Affecting Ionic Selectivity

The selectivity of ion exchangers is subject to the influence of the feed conditions as well as the properties of the exchanger. Both theoretical reasoning and observed results indicate that the ionic selectivity of the exchangers is generally enhanced by the following conditions:

- (1) High degree of cross-linking, i.e., up to about 15%. (Beyond this point the selectivity tends to become constant or decrease.)
- (2) Low concentration of dissolved solids in solution, because the exchanger matrix is highly strained owing to the high swelling pressure. The tendency of the matrix to relax under such conditions makes it prefer a smaller ion (see item 4 below).
- (3) High valence of ion in solution, because the driving force directing counterions toward the ion exchanger is proportional to the ionic charge.
- (4) Ion of small equivalent solvated volume, since smaller ions produce less swelling of the ion exchanger, thereby relieving the stresses of the exchanger matrix.

The influence of temperature on the performance of an ion exchanger, especially the ionic selectivity, must be considered. In general, the selectivity for univalent cations diminishes with increasing temperature, while the reverse may be true for more highly charged cations. In some cases, the selectivity between ions can be reversed by temperature change.²⁴

For dilute aqueous solutions at ordinary temperatures, the empirical rules for ionic selectivity listed in Table D-1 are widely used.

D.3.6 Stability

Because ion exchangers are often employed in cyclic processes (i.e., with repeated exhaustion, regeneration, and/or elution), the physical, chemical, and radiochemical stability of the exchanger material is an important factor.

Physical stability is the capability of the resin to resist breakage and attrition by mechanical forces and other means. Causes of breakage and attrition can be (1) frequent agitation by pumping, etc.; (2) exceedingly high flow rates; and (3) the osmotic shock and an abrupt change in volume resulting from a rapid change in the solution concentration.

Chemical stability is the ability of the ion exchanger matrix to retain its capacity and to resist degradation processes in the chemical environment of the solution being treated. A rapid degradation of the resin matrix can be expected in solutions containing strong oxidants such as hot HNO_3 or chromic acid. The presence of oxygen, together with certain metallic ions (e.g., Cu, Fe, Mn), in appreciable concentrations could slowly degrade the resin by catalytic reactions.

Radiochemical stability is similar to chemical stability, but losses of the exchange capacity and degradation of the exchanger take place under the influence of ionizing radiation. In general, attack by radiation is manifested primarily in the rupture of cross-linkages of strong-acid cation resins and in the degradation of functional groups in anion resins.

Table D-1. Empirical Rules for Ionic Selectivity

1. For counterions of the same valences, the selectivity of the general-purpose ion exchangers increases with increasing atomic number in the same periodic group:

2. The selectivity among ions of the same valences from different periodic groups does not necessarily follow the atomic number relationship:

3. The selectivity among counterions of different valences favors those with high valences:

4. Exceptions to rule 3 exist, for CrO_4^{2-} :

5. The selectivity for H^+ and OH^- varies with the acidity or basicity of the fixed functional groups in the ion exchange system. With strong-acid resins, H^+ is normally placed between Li^+ and Na^+ . Similarly, with strong-base resins, OH^- generally falls between F^- and H_2PO_4^- .

D.4 Kinetics and Mechanism of Ion Exchange Process

Among the major factors affecting the kinetics of ion exchange are (1) the type and concentration of the functional group, (2) the solvent properties, (3) the resin particle size, (4) the extent of cross-linkage, and (5) temperature.

The mechanism of the ion exchange process can be visualized by considering a single spherical ion exchanger particle (strong-acid, H^+ form) depicted in Fig. D-6. Key steps involved in the process are: (1) diffusion of counterion A^+ in the bulk solution through a liquid or layer next to the resin particle; (2) further diffusion of A^+ through the pores to the exchange site; (3) exchange reaction between counterions A^+ and H^+ at the exchange sites; (4) diffusion of counterion H^+ through the pores; and (5) diffusion of H^+ ion through the liquid film to the bulk solution.

The rate of the exchange reaction, step 3, in aqueous solution is generally so high that the controlling step in the overall process is diffusion mass transfer either through the solution film (e.g., at low ionic concentration) or through the resin particle (e.g., at high ionic concentration).

A parameter consisting of ionic concentrations, diffusion coefficients, ionic selectivity (separation factor), etc., has been derived in ref. 12 to evaluate the rate-controlling step as illustrated below. For film diffusion control:

$$\frac{\overline{RD\delta}}{C_A D_r} (5 + 2\alpha_A^H) \gg 1 ; \quad (D-20a)$$

for particle diffusion control:
$$\frac{\overline{RD\delta}}{C_A D_r} (5 + 2\alpha_A^H) \ll 1 , \quad (D-20b)$$

where

$\overline{R}$ = concentration of fixed ionic groups in resin (meq per milliliter of wet resin),

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Fig. D-6. Mechanism of Ion Exchange Process (Cation Exchange Resin, H^+ Form).

C_A = concentration of A in bulk solution (meq/ml),

$\bar{D}$ = interdiffusion coefficient in the liquid film,

δ = thickness of liquid film,

r = radius of resin particle, and

α_A^H = separation factor; see Eq. (D-18).

The instantaneous rate of the overall ion exchange process, γ , may be approximated by the following empirical equations,¹⁵ which assume the linear driving force relation. For particle diffusion control:

$$\gamma = \frac{\bar{D}}{0.071r^2} (\bar{C}_A^* - \bar{C}_A) ; \quad (D-21)$$

for film diffusion control:

$$\gamma = \frac{3\bar{D}}{2\delta r} (C_A - C_A^*) , \quad (D-22)$$

where $\bar{C}_A^*$ and $\bar{C}_A$ are the equilibrium and actual concentrations, respectively, of A in the ion exchanger and C_A^* and C_A are corresponding quantities in the solution.

D.5 Leakage in the Ion Exchange System

Since the ion exchange process is based on the exchange reactions between the counterions in the resins and corresponding counterions in the solution, the contaminants to be removed from the solution must be in ionic form. This implies that nonionized nuclides and nuclides in colloidal particulate form (10^{-3} to 1μ in size) and in form of suspended solids will not be removed by the ion exchange system, although large suspended solids will be partially removed by the ion exchange bed functioning as an expensive filter. The nuclides not removed by ion exchangers constitute

the leakage from the system. Some colloidal particulates can be removed with macroreticular ion exchange resins which have a unique pore structure. These resins have large pore sizes that facilitate access to the inner pores. For details, see ref. 22. A separate filter system is normally installed ahead of the ion exchange unit if appreciable amounts of suspended solids are in the feed.

In addition to the types of leakage mentioned above, small amounts of ionic contaminants always appear in the effluent stream (ion leakage) prior to the breakthrough point. This leakage can occur despite efficient operation of mixed-bed columns, due to factors such as (1) less than complete regeneration in the bottom region of the ion exchange bed, (2) exceedingly high or low feed flow, (3) organic fouling of the ion exchange bed, and (4) channeling in the bed. A brief discussion of these causes and possible remedies is outlined below.

D.5.1 Incomplete Regeneration

Regeneration of the exhausted ion exchange column is usually carried out by the downward flow of the regenerant. Accordingly, the exchanger bed near the top tends to be more completely regenerated than the bottom unless a large excess of regenerant is used. Thus, at the end of the regeneration phase, the bottom section of the bed may still have residual amounts of ionic contaminants from the previous exhaustion phase, depending upon the level of regeneration. When the feed flow of water or waste solution is resumed, the counterions in the freshly regenerated ion exchangers in the top section are replaced by ionic contaminants in the feed. The counterions leaving the top section of the bed, in turn, exchange with the residual ionic contaminants in the bottom section which are then discharged from the bottom as the ion leakage. A similar condition may develop when technical-grade regenerants containing appreciable amounts of impurities are employed.

D.5.2 Exceedingly High or Low Feed Flow

An exceedingly high rate of feed flow can result in a significant ion leakage primarily because of the insufficient residence time allowed for ionic nuclides to be transported from the bulk solution to the exchanger. The ion leakage likewise takes place when the feed rate is too low (< 1 gpm per square foot of cross section of bed), presumably due to the elution (regeneration) effect. Similar behavior has been observed as the ion exchange system was restarted after a pause during the exhaustion phase. This condition is particularly noticeable in the separate-bed system. The ion leakage of the last two types can be minimized by recycling the effluent to the ion exchange unit.

D.5.3 Organic Fouling of Ion Exchange Bed

Organic fouling is encountered with a system containing strong-base anion exchange resins which are capable of undergoing ion exchange with weakly ionized organic acids, as well as absorbing organic acids and other organic matter in the feed. These organic contaminants cannot always be completely removed in the regeneration step with caustic. The accumulation of organic matter causes reduction in the exchange capacity and brings about ion leakage. Organic fouling is more serious with a mixed-bed system and is attributable to the lower pH value of the effluent from the mixed bed. Macroreticular resins have been found useful in removing organic foulants²⁵ and can serve as an effective pretreatment of feed to the main column. Similarly, strong-base anion exchange resins regenerated with NaCl solution can also function as an organic trap or scavenger.²⁶ Regeneration of the strong-base anion resins with salt solution can effectively remove the organics; however, the chloride in the regenerant may pose corrosion problems. Removal of organics and other "crud" by an ultrasonic technique is also used.^{7,20}

**APPENDIX E. LIST OF PUBLISHED LITERATURE OF EFFECTIVENESS
OF ION EXCHANGERS IN REMOVING SPECIFIC NUCLIDES**

<u>Article</u>	<u>Specific Nuclides Involved</u>
E-1 Dickinson, B. N., and Higgins, I. R., "Selective Ion-Exchange Sorption of Cs-137 and I-131 from Borate Solution," <u>Nucl. Sci. Eng.</u> 27: 131-39 (1967).	^{137}Cs , ^{131}I
E-2 Sisson, A. B., Scott, J. S., and Selby, K. A., "Resin Evaluation for Removal of Radionuclides from PWR Steam Generator Blowdowns," presented at International Water Conference of Eng. Soc. of Western Pennsylvania, Nov. 2-4, 1971.	^{137}Cs , ^{131}I , ^{60}Co , ^{99}Mo
E-3 Brooksbank, R. E., Browder, F. N., Holcomb, R. R., and Whitson, W. R., <u>Low-Radioactivity-Level Waste Treatment, Part II</u> , USAEC report ORNL-3349 (May 1963).	^{137}Cs , ^{90}Sr
E-4 Higgins, I. R., and Messing, A. F., <u>Development of a Continuous Ion Exchange Process for the Removal and Recovery of High-Purity Cesium from Alkaline Waste</u> , USAEC report ORNL-2491 (October 1958).	^{137}Cs , ^{90}Sr , ^{144}Ce , Na
E-5 Fletcher, W. D., <u>Ion Exchange in Boric Acid Solutions with Radioactive Decay</u> , USAEC report WCAP-3716 (November 1962).	^{134}Cs , ^{131}I , ^{58}Co , ^{54}Mn , ^{89}Sr
E-6 Lindsay, W. T., Jr., and Abrams, C. S., <u>Ion Exchange Removal of Fission Products from High Purity Water - Development Work Associated with the PWR Radioactive Waste Disposal System</u> , USAEC report WAPD-PWR-CP-2636 (1957).	^{137}Cs , ^{131}I , ^{140}Ba , ^{144}Ce , ^{90}Sr , ^{99}Mo , ^{106}Ru , ^{95}Zr , etc.
E-7 Gray, J. E., Hamilton, W. H., and Wynne, W. E., eds., <u>Shippingport Operations from Start-Up to First Refueling, Dec. 18, 1957, to Oct. 7, 1959</u> , DICS-364, p. IV-68.	^{83}Br , ^{89}Sr , ^{131}I , ^{136}Cs , ^{140}Ba
E-8 Weisman, J., and Bartnoff, S., <u>The Saxton Chemical Shim Experiment</u> , USAEC report WCAP-3269-24 (July 1965).	^{137}Cs , ^{131}I , ^{58}Co , ^{60}Co

E-9	Holcomb, R. R., <u>Low-Radioactivity Level Waste Treatment, Part I</u> , USAEC report ORNL-3322 (June 1963).	^{137}Cs , ^{106}Ru ,
^{60}Co , $^{95}\text{Zr-Nb}$,
^{65}Zn , ^{90}Sr
E-10	Roberts, J. T., and Holcomb, R. R., <u>A Phenolic Ion Exchange Process for Decontaminating Low-Radioactivity-Level Process Water Wastes</u> , USAEC report ORNL-3036 (May 1961).	^{137}Cs , ^{90}Sr
E-11	King, L. J., and Ichikawa, M., <u>Pilot Plant Demonstration of the Decontamination of Low-Level Process Wastes by a Recycle Scavenging-Precipitation Ion Exchange Process</u> , USAEC report ORNL-3863 (December 1965).	^{137}Cs , ^{90}Sr
E-12	Yee, W. C., DeLora, F., and Shockley, W. E., <u>Low-Level Radioactive Waste Treatment</u> , USAEC report ORNL-4472 (July 1970).	^{137}Cs , ^{90}Sr , ^{144}Ce
^{131}I , ^{60}Co , ^{106}Ru ,
$^{95}\text{Zr-Nb}$
E-13	Mercer, B. W., <u>Adsorption of Trace Ions from Intermediate Level Radioactive Wastes by Ion Exchange</u> , USAEC report BNWL-180 (January 1966).	^{137}Cs , ^{90}Sr , ^{106}Ru
$^{95}\text{Zr-Nb}$,
$^{144}\text{Ce-Pr}$

APPENDIX F. LIST OF ORGANIZATIONS FURNISHING TECHNICAL DATA

1. Utility companies:

<u>Name</u>	<u>Power plants involved</u>
Commonwealth Edison Company	Dresden 1, 2, and 3; Zion 1 and 2
Consolidated Edison Company	Indian Point 1 and 2
Consumers Power Company	Big Rock Point; Palisades; Midland 1 and 2
Detroit Edison Company	Fermi 2
Long Island Lighting Company	Shoreham 1
Niagara Mohawk Power Corporation	Nine Mile Point
Northern States Power Company	Monticello
Philadelphia Electric Company	Peach Bottom 1, 2, and 3; Limerick 1 and 2
Pacific Gas and Electric Company	Humboldt Bay
Rochester Gas and Electric Corp.	Ginna
Southern California Edison Company	San Onofre 1, 2, and 3
Tennessee Valley Authority	Browns Ferry 1, 2, and 3
Wisconsin Michigan Power Company	Point Beach 1 and 2
Wisconsin Electric Power Company	
Yankee Atomic Electric Company	Yankee-Rowe

2. Demineralizer and ion exchange resin manufacturers:

Crane Company (Cochrane Div.); Diamond Shamrock Chemical Company;
Dow Chemical Company; Graver Water Conditioning Company; Ionac
Chemical Company; Ionics, Inc.; Permutit Company; Rohm & Haas
Company

3. Nuclear-steam-supply-system vendors and architect-engineers:

Babcock & Wilcox Company; Combustion Engineering, Inc.; General
Electric Company; Stone & Webster Engineering Corporation; West-
inghouse Electric Corporation

4. Other organizations:

Allied Chemical Company, National Reactor Testing Station (Idaho
Falls); U.S. Atomic Energy Commission, Pittsburgh Naval Reactors
Office (Shippingport Reactor Installation)

**APPENDIX G. EQUIPMENT SPECIFICATIONS AND PERFORMANCE
OF ION EXCHANGE SYSTEMS IN BWR POWER PLANTS**

Table G-1. Equipment Specifications and Performance

Plant Name	Installed Power [MW(t)]	Ion Exchange System Technical Data				
		Resin Type ^c	Function (Treatment Of)	No. of Units	Dimension and/or Resin Volume	Regeneration
Big Rock Point ^b	240	N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Reactor coolant	1	24 ft ³	No
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Condensate	3 (1 spare)	54 ft ³ ea.	No
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Radwaste (liq.)	1	24 ft ³	No
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Makeup water	1	72 ft ³	Yes
Browns Ferry 1, 2, and 3	3,293 ea.	Powdex	Fuel pool water	2/reactor	1/4 in. thick x 250 ft ² ; 6 ft ³ /ea.	No
		Powdex	Condensate	9/reactor	1/4 in. thick x 800 ft ² ; 17 ft ³ /ea.	No
		N.G.-M.B.	Radwaste (liq.)	1	7.5 ft diam x 3 ft deep, 125 ft ³	No
		Powdex	Reactor coolant	2	120 ft ² , 2.5 ft ³	No
Dresden 1 ^b	700	N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Reactor coolant	2	5 ft diam x 3.8 ft deep, 75 ft ³	No
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Condensate	4 (1 spare)	6.5 ft diam x 5.25 ft deep, 120 ft ³	Yes
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Radwaste (high purity)	1	5 ft diam x 4.6 ft deep, 90 ft ³	No
Dresden 2 ^b and 3 ^b	2,527 ea.	N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Fuel pool water	1/reactor	7.5 ft diam x 4.1 ft deep, 180 ft ³	Yes
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Reactor coolant	1/reactor	7.5 ft diam x 4.1 ft deep, 180 ft ³	Yes
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Condensate	7 (1 spare)	6-ft-diam sphere	Yes
		N.G.-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Radwaste (liq.)	1	7.5 ft diam x 4.1 ft deep, 180 ft ³	Yes

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Performance of Ion Exchange Systems in BWR Power Plants^a

n	Avg. Service Life ^d	Approximate Feed Conditions ^e			Performance Measurement		DF ^f Assumed or Used
		Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured	
	3.5 months	90			Radioactivity	I-activity: gross β+γ of filt.; gross β+γ of crud	3000 20 500
	80 days	50 gpm ft ²	80-90	6.7-6.9	Chem. anal.	Fe and Cu in filt. and crud	
	5-6 weeks (~4 hr/day)	40			Electrical conductivity	Conductivity	<1.2 micro-mho/cm
					Electrical	Conductivity	<1.2 micro-mho/cm
		600/unit	100-120	~7.0	Radioactivity		
		3000/unit	100-120	~7.0	Pressure drop	ΔP across bed	<25 psig
		420		6-8	Radioactivity	Gross β+γ	<10 ⁻⁴ μCi/ml (except ³ H)
		135	100-120	6-8	Pressure drop	ΔP across bed	<25 psig
	~10 months (~1 fuel cycle)	200	~110	~7.0	Radioactivity	Gross β+γ	>100-1000
	~2 months ea.	1600/unit	<130	~7.0	Radioactivity	Gross β+γ	
		200	~110	~7.0	Radioactivity	¹³¹ I; gross β+γ	>100-1000 >100
		<600	<130	~7.0	Electrical conductivity; radioactivity	Conductivity Gross β+γ	<1 micro-mho/cm
	~10 months (~1 fuel cycle)	~600	~110	~7.0	Radioactivity	Gross β+γ (<1 R/hr)	
	~2 months ea.	3300/unit	<130	~7.0	Radioactivity	Gross β+γ (<1 R/hr)	
	~10 days	200	~110	~7.0	Electrical conductivity; radioactivity (spot check); chem. anal.	Conductivity Gross β+γ SiO ₂	<1 micro-mho/cm

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Performance Measurement		DF ^f Assumed or Used	Remarks
Method	Variable Measured		
Radioactivity	I-activity; gross $\beta+\gamma$ of filt.; gross $\beta+\gamma$ of crud	3000 20 500	
Chem. anal.	Fe and Cu in filt. and crud		
Electrical conductivity	Conductivity	<1.2 micro- mho/cm	
Electrical	Conductivity	<1.2 micro- mho/cm	
Radioactivity			
Pressure drop	ΔP across bed	<25 psig	Radioactivity and conductivity also measured.
Radioactivity	Gross $\beta+\gamma$	<10 ⁻⁴ $\mu\text{Ci/ml}$ (except ³ H)	
Pressure drop	ΔP across bed	<25 psig	Radioactivity and conductivity also measured.
Radioactivity	Gross $\beta+\gamma$	>100-1000	Resin replaced at refueling.
Radioactivity	Gross $\beta+\gamma$		Regeneration if >1 R/hr; control- ling nuclides = ¹⁴⁰ Ba- ¹⁴⁰ La.
Radioactivity	¹³¹ I; gross $\beta+\gamma$	>100-1000 >100	
Electrical conductivity; Radioactivity	Conductivity	<1 micro- mho/cm	New installation.
Radioactivity	Gross $\beta+\gamma$		Regeneration at refueling.
Radioactivity	Gross $\beta+\gamma$ (<1 R/hr)		
Radioactivity	Gross $\beta+\gamma$ (<1 R/hr)		
Electrical conductivity; Radioactivity (at check); Chem. anal.	Conductivity	<1 micro- mho/cm	Recycled to plant.
	Gross $\beta+\gamma$		
	SiO ₂		

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Plant Name	Installed Power [MW(t)]	Ion Exchange System Technical Data				Regeneration
		Resin Type ^c	Function (Treatment Of)	No. of Units	Dimension and/or Resin Volume	
Fermi 2	3,293	Powdex	Condensate	7	915 ft ² coverage, 183 lb resin/unit, 745 element (1.1 in. diam x 5 ft long ea.)	No
		N.G.-M.B. (uncertain)	Equipment drain	1	~45 ft ³	No
			Floor drain	1	~45 ft ³	No
Humboldt Bay 3 ^b	240	N.G.-M.B. 2C.(H ⁺) + 1A.(OH ⁻)	Reactor coolant	1	2.5 ft diam x 3.6 ft deep, 17.5 ft ³	No
		M.B. 2C.(H ⁺) + 1A.(OH ⁻)	Condensate	4 (1 spare)	4 ft diam x 3 ft deep, 38 ft ³	Yes
		N.G.-M.B. 2C.(H ⁺) + 1A.(OH ⁻)	Fuel pool water	1	2 ft diam x 4 ft deep	No
Limerick	3,293	Powdex	Reactor coolant	2	158 ft ² coverage, 36 lb resin/unit	No
		M.B. 2C.(H ⁺) + 1A.(OH ⁻)	Equipment drain	1	6 ft diam x 3 ft deep, 85 ft ³	No
		M.B. 1C.(H ⁺) + 2A.(OH ⁻)	Floor drain	1	6 ft diam x 3 ft deep, 85 ft ³	No
Monticello ^b	1,670	Powdex	Reactor coolant	2	93 ft ² coverage per unit	No
		Powdex	Condensate	5	792 ft ² coverage per unit	No
		Powdex + M.B. C.(H ⁺) + A.(OH ⁻)	High purity waste, fuel pool water	1	Powdex: 225 ft ² M.B.: 3.45 ft diam x 3.42 ft deep, 32 ft ³	No

Table G-1 (continued)

Regeneration	Avg. Service Life ^d	Approximate Feed Conditions ^e			Performance Measurement		DF ^f Assumed or Used
		Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured	
No		3.8 gpm/ft ²	94-125	6.5-7.5			
No		140	40-140	6-8	Radioactivity	Gross βγ	100
No		140	40-140	6-8	Radioactivity	Gross βγ	100
No	~1 year (1 fuel cycle)	40	95	7.7	Radioactivity	Iodines Crud (⁶⁰ Co, ⁵⁴ Mn)	10,000 200
Yes	~1 year (1 fuel cycle)	550/unit	115	6.8	Radioactivity	Iodines Crud (⁶⁰ Co, ⁵⁴ Mn)	3,000 50
No		20	65	7.1		Iodines Crud (⁶⁰ Co, ⁵⁴ Mn)	10,000 200
No		180/unit (Design)	120-150	5.5-8.5	Elec. cond.	Conductivity	≤0.1 micro- mho/cm
					Radioactivity	Gross βγ	10-100
No		280	40-140	6-8	Elec. cond.	Conductivity	<1 micro- mho/cm
					Radioactivity	Gross βγ	100
No		280	40-140	6-8	Elec. cond.	Conductivity	<1 micro- mho/cm
					Radioactivity	Gross βγ	100
No	7 days	80/unit	<150	7	Radioactivity	Gross βγ (crud) Gross βγ (filt.) ¹³¹ I ⁶⁴ Cu (cation) ⁶⁴ Cu (anion)	~80-100 40 ~85-90 ~10-40 ~30
No	7-10 days	2000/unit	150	7	Radioactivity		
No	For waste Powdex: 2-3 days M.B.: 3-4 months	100	<100	7	Radioactivity	Crud (⁵⁸ Co, ⁶⁰ Co, ⁶⁴ Cu) ¹³¹ I ⁹⁹ Tc Cs	~700-1000 ~400 ~100 ~4 ~30

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Conditions ^e	Performance Measurement		DF ^f Assumed or Used	Remarks
	pH	Method Variable Measured		
25	6.5-7.5			
40	6-8	Radioactivity	Gross β+γ 100	
40	6-8	Radioactivity	Gross β+γ 100	
	7.7	Radioactivity	Iodines Crud (⁶⁰ Co, ⁵⁴ Mn) 10,000 200	Resin replaced at refueling.
	6.8	Radioactivity	Iodines Crud (⁶⁰ Co, ⁵⁴ Mn) 3,000 50	
	7.1		Iodines Crud (⁶⁰ Co, ⁵⁴ Mn) 10,000 200	Used only occasionally. Radwaste treated by filter and evaporator.
50	5.5-8.5	Elec. cond.	Conductivity ≤0.1 micro- mho/cm	
		Radioactivity	Gross β+γ 10-100	
40	6-8	Elec. cond.	Conductivity <1 micro- mho/cm	
		Radioactivity	Gross β+γ 100	
40	6-8	Elec. cond.	Conductivity <1 micro- mho/cm	
		Radioactivity	Gross β+γ 100	
	7	Radioactivity	Gross β+γ (crud) ~80-100 Gross β+γ (filt.) 40 ¹³¹ I ~85-90 ⁶⁴ Cu (cation) ~10-40 ⁶⁴ Cu (anion) ~30	Replacement of Powdex bed is determined by the total radioactivity of the bed, and not by the DF.
	7	Radioactivity		
	7	Radioactivity	Crud (⁵⁸ Co, ⁶⁰ Co, ⁶⁴ Cu) ~700-1000 ~400 ¹³³ I ~100 ⁹⁹ Mo ~4 Cs ~30	Powdex bed is used for fuel pool water occasionally.

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Plant Name	Installed Power [MW(t)]	Ion Exchange System Technical Data				
		Resin Type ^c	Function (Treatment Of)	No. of Units	Dimension and/o Resin Volume	Regeneration
Nine Mile Point ^b	1,850	Ma-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Reactor coolant	2 (1 spare)	8 ft diam x 3 ft deep, 168 ft ³ /unit	Yes
		Ma-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Condensate	6 (1 spare)	9 ft diam x 3 ft deep, 168 ft ³ /unit	Yes
		Ma-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Equipment drains	1	8 ft diam x 3 ft deep, 168 ft ³ /unit	Yes
Peach Bottom 2 and 3	3,294 ea.	Powdex	Reactor coolant	2/reactor	180 ft ² coverage	No
		Powdex	Condensate	10/reactor (1 spare)	180 ft ² coverage	No
		Powdex	Fuel pool water	3 (1 spare)		No
		H.G.-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Equipment drains	1	3-5 ft deep, 85 ft ³	No
Shoreham 1	2,550	H.G.-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Fuel pool water	1	3-4 ft deep	Yes (mayb)
		Powdex	Reactor coolant	2	126 ft ² coverage	No
		Ma-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Condensate	8 (1 spare)	8.8 ft diam x 4 ft deep, 250 ft ³	Yes
		Ma-M.B., 2C.(H ⁺) + 1A.(OH ⁻)	Radwaste	2	50 ft ³	No

^aAs of summer 1972.

^bOperating plant.

^cH.G. = nuclear grade; M.B. = mixed bed; 2C.(H⁺) + 1A.(OH⁻) = 2 vol. strong acid cation resins (H⁺ form) + 1 vol. anion resins (OH⁻ form); Ma = macroreticular resins.

^dBetween regeneration or replacement of resins.

^eFeed conditions for plants under construction represent the design basis.

^fDF = (radioactivity in feed/radioactivity in effluent). Other criteria used are also indicated.

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Table G-1 (continued)

Regeneration	Avg. Service Life ^d	Approximate Feed Conditions ^e			Performance Measurement		DF ^f Assumed or Used
		Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured	
Yes	>6 months	4 gpm/ft ²	95	7	Radioactivity	¹³¹ I- ¹³⁵ I Gross β+γ of filt. and crud	100
					Chem. anal.	Si	
Yes	2-3 months	52 gpm/ft ²	95	7	Radioactivity	Gross β+γ of filt. and crud	100
Yes	2-3 weeks	15 gpm/ft ²	95	7	Radioactivity	Gross β+γ	
No		280 (Design)	40-140	6-8	Elec. cond.	Conductivity	<0.1 micro- mho/cm
					Radioactivity	Gross β+γ	10-100
No		3000/unit	40-140	6-8	Elec. cond.	Conductivity	<0.1 micro- mho/cm
					Radioactivity	Gross β+γ	10-100
No		450/unit	<150	6-8	Elec. cond.	Conductivity	<1 micro- mho/cm
					Radioactivity	Gross β+γ	100
No		280 (Design)	40-140	6-8	Elec. cond.	Conductivity	<1 micro- mho/cm
					Radioactivity	Gross β+γ	100
Yes (maybe)		240	<125	6-8	Radioactivity		
No		100/unit	<150	6-8	Radioactivity		
Yes		1500-3400/unit	100-130	6-8	Radioactivity	Gross β+γ (crud)	100
No		150	100-130	6-8	Radioactivity	Gross β+γ, Cs	100

a) + 1 vol. of strong base

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Feed Conditions ^e		Performance Measurement		DF ^f Assumed or Used	Remarks
Temp. (°F)	pH	Method	Variable Measured		
95	7	Radioactivity	^{131}I ~ ^{135}I Gross $\beta+\gamma$ of filt. and crud	100	Si determines exhaustion of resin.
		Chem. anal.	Si		
95	7	Radioactivity	Gross $\beta+\gamma$ of filt. and crud	100	
95	7	Radioactivity	Gross $\beta+\gamma$		Activities of individual nuclides determ. occasionally.
40-140	6-8	Elec. cond.	Conductivity	<0.1 micro- mho/cm	
		Radioactivity	Gross $\beta+\gamma$	10-100	
40-140	6-8	Elec. cond.	Conductivity	<0.1 micro- mho/cm	
		Radioactivity	Gross $\beta+\gamma$	10-100	
<150	6-8	Elec. cond.	Conductivity	<1 micro- mho/cm	
		Radioactivity	Gross $\beta+\gamma$	100	
40-140	6-8	Elec. cond.	Conductivity	<1 micro- mho/cm	
		Radioactivity	Gross $\beta+\gamma$	100	
<125	6-8	Radioactivity			
<150	6-8	Radioactivity			
100-130	6-8	Radioactivity	Gross $\beta+\gamma$ (crud)	100	Si determines exhaustion of resin.
100-130	6-8	Radioactivity	Gross $\beta+\gamma$, Cs	100	

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**APPENDIX H. EQUIPMENT SPECIFICATIONS AND PERFORMANCE
OF ION EXCHANGE SYSTEMS IN PWR POWER PLANTS**

Table H-1. Equipment Specifics

Plant Name	Installed Power [MW(t)]	Ion Exchange System Technical Data				Regeneration
		Resin Type ^c	Function (Treatment Of)	No. of Units	Dimension and/or Resin Volume	
Ginna ^b	1,300	N.G.-M.B., 1C.(⁷ Li ⁺) + 1A.(OH ⁻)	Primary loop coolant	2 (1 spare)	12.5 ft ³	1
		N.G.-C.(⁷ Li ⁺)	Same	1	12.5 ft ³	1
		N.G.-A.(OH ⁻)	Deborating	2	30 ft ³	1
		N.G.-M.B., 1C.(⁷ Li ⁺) + 1A.(OH ⁻)	Reuse water (letdown)	1	12.5 ft ³	1
		N.G.-C.(H ⁺)	Same	2 (1 spare)	12.5 ft ³	1
		N.G.-A.(OH ⁻)	Evaporator condensate	2 (1 spare)	12 ft ³	1
		N.G.-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Radwaste	2 (1 spare)	1 ft ³	1
Indian Point 1 ^b	615	N.G.-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Primary loop coolant	3	4.5 ft diam x 5.5 ft deep, 63 ft ³ /unit	1
		N.G.-C.(H ⁺)	Same	3	4.5 ft diam x 5.5 ft deep, 63 ft ³ /unit	1
		N.G.-A.(OH ⁻)	Deborating	2	4 ft diam x 6 ft deep, 60 ft ³ /unit	1
		N.G.-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Waste evap. condensate	1	1.7 ft diam x 5.5 ft deep, 8 ft ³	1
Indian Point 2	2,758	N.G.-M.B., 1C.(⁷ Li ⁺) + 1A.(OH ⁻)	Primary loop coolant	2	2.7 ft diam x 7.5 ft deep, 30 ft ³ /unit	1
		N.G.-C.(H ⁺)	Letdown stream	1	1.7 ft diam x 7.8 ft deep, 12 ft ³	1
		N.G.-C.(H ⁺)	B recycle system	4	1.7 ft diam x 7.8 ft deep, 12 ft ³	1
		N.G.-A.(OH ⁻)	Letdown stream deborating	2	2.7 ft diam x 7.5 ft deep, 30 ft ³ /unit	1
		N.G.-A.(OH ⁻)	B recycle evap. condensate	2	1.7 ft diam x 7.8 ft deep, 12 ft ³ /unit	1

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Specifications and Performance of Ion Exchange Systems in PWR Power Plants^a

Regeneration	Avg. Service Life ^d	Approximate Feed Conditions ^c			Performance Measurement		DF ^f Assumed or Used
		Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured	
No	6-9 months	40-60	140	4.2-10.5	Radioactivity	Gross β+γ	10
No	2-3 years	40-60	140	4.2-10.5	Radioactivity	Gross β+γ	100
Yes		40-60	140	4.2-10.5			
No		12	90	4.2-10.5	Radioactivity Elec. cond.	Gross β+γ Conductivity	
No	6-8 hr or 2-3 weeks	12	90	4.2-10.5	Elec. cond.	Conductivity	
Yes		12	90	6.0-8.0	Elec. cond.	Conductivity	
No		2	90	6.5-8.5	Radioactivity	Gross β+γ	20
No		Avg. 80/unit Max. 160/unit	90-100	5-6			
No		Avg. 80/unit Max. 160/unit	90-100	5-6			
No		80/unit	90-100	5-6			
No		12		6.5			
No		27-109/unit	110-135	5-10.5			
No		10-42	110-135	5-10.5			
No		10-42	<100	5-10.5			
Yes		27-109/unit	120	7-10.5			
Yes		10-42/unit	120	6-8			

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Systems in PWR Power Plants^a

Feed Conditions ^c		Performance Measurement		DF ^f Assumed or Used	Remarks
Temp. (°F)	pH	Method	Variable Measured		
140	4.2-10.5	Radioactivity	Gross $\beta+\gamma$	10	Conductivity at selected location.
140	4.2-10.5	Radioactivity	Gross $\beta+\gamma$		24 hr/month use only.
140	4.2-10.5				New installation.
90	4.2-10.5	Radioactivity Elec. cond.	Gross $\beta+\gamma$ Conductivity	100	Used only a few days per month.
90	4.2-10.5	Elec. cond.	Conductivity		
90	6.0-8.0	Elec. cond.	Conductivity		
90	6.5-8.5	Radioactivity	Gross $\beta+\gamma$	20	
90-100	5-6				
90-100	5-6				
90-100	5-6				pH 6-10 with ⁷ Li addition.
	6.5				
110-135	5-10.5				
110-135	5-10.5				For Cs removal.
<100	5-10.5				For Li removal.
120	7-10.5				
120	6-8				For H ₃ BO ₃ removal.

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Plant Name	Installed Power [MW(t)]	Ion Exchange System Technical Data			
		Resin Type ^c	Function (Treatment Of)	No. of Units	Dimension and/or Resin Volume
Midland	2,452 ea.	N.G.-M.B.	Primary loop coolant	2	4 ft diam x 4 ft deep, 50 ft ³ /unit
		N.G.-A.(OH ⁻)	Deborating	2	4 ft diam x 4 ft deep, 50 ft ³ /unit
		N.G.-M.B.	High purity waste	3	4 ft diam x 4 ft deep, 50 ft ³ /unit
		N.G.-M.B.	High purity waste evap. condensate	1	4 ft diam x 4 ft deep, 50 ft ³ /unit
		N.G.-M.B.	Low purity waste	1	4 ft diam x 4 ft deep, 50 ft ³ /unit
		N.G.-M.B.	Low purity waste evap. condensate	1	4 ft diam x 4 ft deep, 50 ft ³ /unit
Palisades ^b	2,212	N.G.-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Primary loop coolant	2	3.5 ft diam x 3.25 deep, 32 ft ³ /unit
		N.G.-A.(OH ⁻)	Deborating	1	3.5 ft diam x 3.25 deep, 32 ft ³ /unit
		N.G.-M.B., C.(H ⁺) + A*(OH ⁻)	High purity waste	3	3.5 ft diam x 3.25 deep, 32 ft ³ /unit
Point Beach 1 ^b and 2	1,518 ea.	N.G.-M.B., C.(⁷ Li ⁺) + A.(OH ⁻)	Primary loop coolant	2/reactor	2.1 ft diam x 5.7 ft deep, 20 ft ³ /unit
		N.G.-A.(OH ⁻)	Deborating	2/reactor	2.6 ft diam x 5.7 ft deep, 30 ft ³ /unit
		N.G.-C.(H ⁺)	Primary loop coolant (Li removal)	1/reactor	1.6 ft diam x 6 ft deep, 12 ft ³
		N.G.-C.(H ⁺)	B recycle evap. feed	4	1.6 ft diam x 6 ft deep, 12 ft ³ /unit
		N.G.-A.(OH ⁻)	B recycle evap. condensate	3	1.6 ft diam x 6 ft deep, 12 ft ³ /unit
		N.G.-M.B., C.(H ⁺) + A.(OH ⁻)	Waste evap. condensate	2 (1 spare)	2.6 ft diam x 5.7 ft deep, 30 ft ³

Table E-1 (continued)

/or e	Regen- eration	Avg. Service Life ^d	Approximate Feed Conditions ^e			Performance Measurement		As or
			Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured	
ft /unit	No		150/unit	120	5.5-7.0			
ft /unit	No		150/unit	70-100	5.5-6.5			
ft /unit	No		150/unit	70-120	5.5-6.5			
ft /unit	No		150/unit	120	7.0-8.0			
ft /unit	No		50	70-100	6.0-9.0			
ft /unit	No		50	70-120	7.0-8.0			
0.25 ft unit	No		40-160/unit	<140	4.5-8.0			
0.25 ft unit	No		40-160/unit	<140	4.5-8.0			
0.25 ft unit	No		48-100/unit	80				
0.7 ft unit	No	6 months	40/unit	70-140	6.0-7.0	Radioactivity	Iodines Gross β+γ	
0.7 ft unit	Yes		40/unit	70-140	5.0-7.5	Radioactivity	Iodines Other anions	
0 ft	No		40/unit	70-140	6.0-7.0	Radioactivity Chem. anal.	Cs, Y, Mo Li	
0 ft unit	No		12.5/unit	70-140	5.0-7.5	Chem. anal.	Li	
0 ft unit	No		12.5/unit	70-140	5.0-7.5	Radioactivity Chem. anal.	Iodines B	
0.7 ft	No		35/unit	70-140	4.0-7.0	Elec. cond.	Conductivity	

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itions ^e	Performance Measurement		DF ^f Assumed or Used	Remarks
	Method	Variable Measured		
5.5-7.0				
5.5-6.5				
5.5-6.5				
7.0-8.0				
6.0-9.0				
7.0-8.0				
4.5-8.0				
4.5-8.0				
6.0-7.0	Radioactivity	Iodines Gross $\beta+\gamma$	10	No appreciable change.
5.0-7.5	Radioactivity	Iodines Other anions	10 8-10	
6.0-7.0	Radioactivity Chem. anal.	Cs, Y, Mo Li	10	
5.0-7.5	Chem. anal.	Li		
5.0-7.5	Radioactivity Chem. anal.	Iodines B	10	For H ₃ BO ₃ removal.
4.0-7.0	Elec. cond.	Conductivity		New installation.

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Plant Name	Installed Power [MW(t)]	Ion Exchange System Technical Data				Reg era
		Resin Type ^c	Function (Treatment Of)	No. of Units	Dimension and/or Resin Volume	
San Onofre 1 ^b	1,347	N.G.-M.B., C.(Li ⁺) + A.(OH ⁻)	Primary loop coolant	2	2.4 ft diam x 5.5 ft deep, 25 ft ³	N
		N.G.-C.(H ⁺)	Primary loop coolant (Li removal)	1	1.6 ft diam x 5.5 ft deep, 11 ft ³	N
		N.G.-M.B., C.(H ⁺) + A.(OH ⁻)	Radwaste	4	2.4 ft diam x 5.5 ft deep, 25 ft ³	N
Sequoyah 1 and 2	3,423 ea.	N.G.-M.B., C.(Li ⁺) + A.(OH ⁻)	Primary loop coolant	2/reactor	3.6 ft diam x 3 ft deep, 30 ft ³ /unit	N
		N.G.-C.(H ⁺)	Primary loop coolant (Cs and Li removal)	1/reactor	2.5 ft diam x 3 ft deep, 20 ft ³	N
		N.G.-C.(H ⁺)	B recovery evap. feed	4	3.6 ft diam x 3 ft deep, 30 ft ³ /unit	N
		N.G.-A.(OH ⁻)	B recovery evap. condensate	2	3.6 ft diam x 3 ft deep, 30 ft ³ /unit	Y
		N.G.-M.B., C.(H ⁺) + A.(OH ⁻)	Waste evap. condensate	1	3.6 ft diam x 3 ft deep, 30 ft ³ /unit	N
		N.G.-M.B., C.(H ⁺) + A.(OH ⁻)	Fuel pool water	1	3.6 ft diam x 3 ft deep, 30 ft ³ /unit	N
Shippingport ^b	505	N.G.-M.B., C.(NH ₄ ⁺) + A.(OH ⁻)	Primary loop coolant	2	2.6 ft diam x 4.4 ft deep, 23 ft ³	N
		N.G.-M.B., C.(H ⁺) + A.(OH ⁻)	Radwaste	4	2 ft diam x 3 ft deep, 10 ft ³	N

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Table H-1 (continued)

Regeneration	Avg. Service Life ^d	Approximate Feed Conditions ^e			Performance Measurement		DF ^f Assumed or Used
		Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured	
No	3-5 months	90/unit (Design)	250 (Design)	4.5-10.2	Radioactivity	Gross β+γ	10
No		90/unit (Design)	250 (Design)	4.5-10.2	Radioactivity	Gross β+γ	10
No	3 months	100/unit (Design)			Radioactivity	Gross β+γ	>10
No		120/unit	127	4.2-10.5	Radioactivity	¹³¹ I	10
No		75	127	4.2-10.5	Radioactivity	¹³⁴ Cs, ¹³⁶ Cs, ¹³⁷ Cs, ⁹⁰ Y, ⁹¹ Y, ⁹⁹ Mo	10
No		30/unit	120	4.5-10.5			
Yes		30/unit	120	7			
No		30/unit	120	7			
No		100	120	4.5-10.5	Radioactivity	Gross β+γ	10
No		66/unit	120	10.1-10.3	Radioactivity	Gross β+γ	>10-25 200-300
No		10-15/unit	<130	≤7	Radioactivity	Gross β+γ	>20

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Conditions ^e	Performance Measurement		DF ^f Assumed or Used	Remarks
pH	Method	Variable Measured		
4.5-10.2	Radioactivity	Gross $\beta+\gamma$	10	Selected nuclides also analyzed.
4.5-10.2	Radioactivity	Gross $\beta+\gamma$	10	Selected nuclides also analyzed.
	Radioactivity	Gross $\beta+\gamma$	>10	Selected nuclides also analyzed.
4.2-10.5	Radioactivity	^{131}I	10	
4.2-10.5	Radioactivity	^{134}Cs , ^{136}Cs , ^{137}Cs , ^{90}Y , ^{91}Y , ^{99}Mo	10	
4.5-10.5				
7				For H_3BO_3 removal.
7				
4.5-10.5	Radioactivity	Gross $\beta+\gamma$	10	
10.1-10.3	Radioactivity	Gross $\beta+\gamma$	>10-25 200-300	Depending on power level. Normal range. Major nuclides = ^{18}F , ^{58}Co and ^{60}Co .
≤7	Radioactivity	Gross $\beta+\gamma$	>20	Major nuclides = ^{58}Co and ^{60}Co .

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Plant Name	Installed Power [MW(t)]	Ion Exchange System Technical Data			
		Resin Type ^c	Function (Treatment Of)	No. of Units	Dimension and/ Resin Volume
Yankee-Rowe ^b	600	N.G.-M.B., 1C.(H ⁺) + 1A.(OH ⁻)	Primary loop coolant (3 used alternately, one at a time)	1	2.5 ft diam x 4. deep, 20 ft ³
		N.G.-C.(H ⁺)		1	2.5 ft diam x 4. deep, 20 ft ³
		N.G.-A.(OH ⁻)		1	2.5 ft diam x 4. deep, 20 ft ³
		N.G.-M.B., C.(H ⁺) + A.(OH ⁻)	Fuel pool water	1	2.5 ft diam x 4. deep, 20 ft ³
Zion 1 and 2	3,391 ea.	N.G.-M.B., 1C.(⁷ Li ⁺) + 1A.(OH ⁻)	Primary loop coolant	2	2.6 ft diam x 5. deep, 30 ft ³ /unit
		N.G.-C.(H ⁺)	Primary loop coolant (Li removal)	1	2.2 ft diam x 5. deep, 20 ft ³
		N.G.-A.(OH ⁻)	Deborating	1	3.1 ft diam x 5. deep, 43 ft ³
		N.G.-C.* (H ⁺)	Steam generator blowdown	2	5 ft diam x 3.5 deep, 68 ft ³ /unit
		N.G.-A.(OH ⁻)		2	7.5 ft diam x 2.8 deep, 125 ft ³ /unit
		N.G.-M.B.	Waste evap. condensate	1	

^aAs of summer 1972.

^bOperating plant.

^cN.G. = nuclear grade; M.B. = mixed bed; 1C.(H⁺) = 1 vol. of strong acid cation resins in H⁺ form; 1A.
1C.(⁷Li⁺) = 1 vol. of strong acid cation resins in ⁷Li⁺ form; (NH₄⁺) = NH₄⁺ form; A.* = intermediate (phenolic-base carboxylic).

^dBetween regeneration or replacement of resins.

^eFeed conditions for nonoperating plants (i.e., under construction) represent the design basis.

^fDF = (Radioactivity in feed/radioactivity in effluent). Other criteria used are also indicated.

Table H-1 (continued)

Data			Approximate Feed Conditions ^e			Performance Measurement	
ion and/or Volume	Regen- eration	Avg. Service Life ^d	Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured
iam x 4.2 ft 0 ft ³	No	10 months	30-50	70-115	4.5-10.5	Radioactivity	Gross β+γ
iam x 4.2 ft 0 ft ³	No		30-50	70-115	4.5-10.5		Specific activ.
iam x 4.2 ft 0 ft ³	No		30-50	70-115	4.5-10.5	Elec. cond.	Conductivity
iam x 4.2 ft 0 ft ³	No		30-50	70-115	4.5-10.5	Chem. anal.	B, NH ₄ ⁺ , Cl
iam x 4.2 ft 0 ft ³	No		30-50	70-115	4.5-10.5	Radioactivity	
iam x 5.5 ft 0 ft ³ /unit	No		120	127	4.2-10.5		
iam x 5.5 ft 0 ft ³	No		72	127	4.2-10.5		
iam x 5.5 ft 3 ft ³	Yes		120	127	4.2-10.5		
iam x 3.5 ft 58 ft ³ /unit	Yes		125	110	9.5		¹³⁷ Cs, ⁶⁰ Co
iam x 2.8 ft 25 ft ³ /unit	No		125	110	9.5	Radioactivity	¹³¹ I

form; 1A.(OH⁻) = 1 vol. of strong base anion resins in OH⁻ form;
intermediate base anion resins; C.* = weak acid cation resins

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ted.

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Approximate Feed Conditions ^e			Performance Measurement		DF ^f	Remarks
Flow Rate (gpm)	Temp. (°F)	pH	Method	Variable Measured	Assumed or Used	
30-50	70-115	4.5-10.5	Radioactivity	Gross $\beta+\gamma$	5-6 (low pH) 300 (high pH)	
30-50	70-115	4.5-10.5		Specific activ.		
30-50	70-115	4.5-10.5	Elec. cond.	Conductivity	<2 micromho/cm (in absence of B)	
			Chem. anal.	B, NH_4^+ , Cl		
30-50	70-115	4.5-10.5	Radioactivity			
120	127	4.2-10.5				
72	127	4.2-10.5				
120	127	4.2-10.5				
125	110	9.5		^{137}Cs , ^{60}Co	10	
125	110	9.5	Radioactivity	^{131}I	100	

anion resins in OH^- form;
acid cation resins

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APPENDIX I. OAK RIDGE NATIONAL LABORATORY (ORNL) STUDIES IN
SMALL-SCALE EQUIPMENT ON THE DECONTAMINATION OF
RADIOACTIVE LIQUIDS BY ION EXCHANGE

The use of an ion exchange unit has been investigated for the removal of trace quantities of ^{137}Cs , ^{131}I , and Sr in macroquantities of nonradioactive NaCl, Na_2HPO_4 , H_3BO_3 , and LiOH contained in simulated nuclear power plant aqueous streams. Operation of the experimental equipment involved continuous introduction of feed into cation and anion columns placed in series or mixed-bed columns in series. Breakthrough curves were determined by analysis of the effluents from both columns. Radiometric techniques were used to monitor and measure concentrations of ^{137}Cs , ^{131}I , and ^{85}Sr . The experimental parameters and compositions of the 3 types of feed studied are summarized in Table I-1, and the experimental results are summarized in Figs. I-1 through I-7. The findings of the study are summarized in the following sections.

I.1 Effects of NaCl Concentration and Flow Rate

Figure I-1 shows the relatively sharp breakthrough curves exhibited by both Na^+ and Cl^- ions in type 1 feed. The breakthrough of Cl^- ion before that of Na^+ ion is primarily attributable to the lower exchange capacity of the anion resin per unit volume (~ 0.8 meq/ml wet vs ~ 1.9 meq/ml wet for cation resin). The breakthrough of $^{137}\text{Cs}^+$ and $^{131}\text{I}^-$ ions is much slower than that of either Na^+ or Cl^- . A fractional breakthrough of 0.1 required approximately 55 and 71 bed volumes (b.v.) of feed throughput, respectively, for Cl^- and Na^+ , whereas approximately 121 and 130 b.v. were required for $^{137}\text{Cs}^+$ and $^{131}\text{I}^-$, respectively. The empirical rules for ionic selectivity hold in this case even though the concentration of Na^+ was more than 10^6 times that of $^{137}\text{Cs}^+$. The same statement can be made for the selectivity between Cl^- and $^{131}\text{I}^-$ ($\sim 10^{-4}$ ppb). Although the effluent was not analyzed for strontium (nonradioactive in this case), it is expected that it would be taken up by the resin in preference to Na^+ and $^{137}\text{Cs}^+$.

Table I-1. Summary of Experimental Parameters^a
(ORNL Experiments)

Ion Exchange System Equipment Specification

Resin type: polystyrene matrix

- (a) Separate-bed: 1-1/4 vol. strong acid cation resin (H^+ form)
1 vol. strong base anion resin (OH^- form)
- (b) Mixed-bed: 1 vol. strong acid cation resin (H^+ form) +
1 vol. strong base anion resin (OH^- form),
2 columns in series
- (c) Total exchange capacity: Cation resin = 1.9 meq/ml,
anion resin = 0.78 meq/ml

Column dimension: 1 in. diam x 3 ft high

Initial resin bed height:

- (a) Separate-bed: 12-15 in.
- (b) Mixed-bed: 27-28 in.

Feed Conditions

Flow Rate: 4.85 and 9.70 gpm per ft^2 of cross-sectional area of
resin bed

Temperature and pH: ambient ($25^\circ C$), pH = 4.0-7.9

Compositions:

- (a) Type 1 feed: 1000 ppm NaCl, 1.0×10^{-2} $\mu Ci/ml$ ^{137}Cs , 5.9×10^{-3} $\mu Ci/ml$ ^{131}I , 7×10^{-8} mg/ml Sr
 - (b) Type 2 feed: 2500 ppm NaCl, 1.2×10^{-2} or 2.7×10^{-2} $\mu Ci/ml$ ^{137}Cs , 5.2×10^{-3} or 1.2×10^{-2} $\mu Ci/ml$ ^{131}I ,
 7×10^{-8} mg/ml Sr
 - (c) Type 3 feed: 1000 ppm Na_2HPO_4 , 1700 ppm H_3BO_3 , 7 ppm LiOH,
 1.6×10^{-2} or 2.8×10^{-2} $\mu Ci/ml$ ^{137}Cs , 9.8×10^{-3} $\mu Ci/ml$ ^{85}Sr , 6.1×10^{-3} or 1.4×10^{-2} $\mu Ci/ml$ ^{131}I
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Fig. I-1. Breakthrough Behavior of Various Ionic Species in a Separate-Bed Ion Exchange System for Treatment of Type 1 Feed (Cation Column Followed by Anion Column).

Fig. I-2. Breakthrough Behavior of Various Ionic Species in a Separate-Bed Ion Exchange System for Treatment of Type 2 Feed (Cation Column Followed by Anion Column).

Fig. I-3. Breakthrough Behavior of Various Ionic Species in a Separate-Bed Ion Exchange System for Treatment of Type 3 Feed (Cation Column Followed by Anion Column).

Fig. I-4. Breakthrough Behavior of Various Ionic Species in Mixed-Bed Ion Exchange System for Treatment of Type 3 Feed (Column A). Arrangement: Two mixed-bed columns, Column A followed by Column B.

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Fig. I-5. Breakthrough Behavior of Various Species in Mixed-Bed Ion Exchange System for Treatment of Type 3 Feed (Column B). Arrangement: Two mixed-bed columns, Column B followed by Column A.

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Fig. I-6. Specific Electrical Conductivity and pH of Effluents from a Separate-Bed Ion Exchange System for Treatment of Type 1 Feed (Cation Column Followed by Anion Column).

Fig. I-7. Specific Electrical Conductivity and pH of Effluents for Mixed-Bed Ion Exchange System for Treatment of Type 3 Feed (Two Mixed-Bed Columns, Column A Followed by Column B).

When the concentrations of Na^+ and Cl^- were increased by a factor of 2.5 in type 2 feed, the breakthrough for Na^+ , Cl^- , and $^{137}\text{Cs}^+$ occurred sooner (e.g., ~ 25 b.v. for Na^+ and Cl^- , and ~ 50 b.v. for $^{137}\text{Cs}^+$ at 0.1 fractional breakthrough or $\text{DF} = 10$) as shown by the broken curves in Fig. I-2. The effect on $^{131}\text{I}^-$ breakthrough is slight. The higher salt concentration also produced somewhat steeper breakthrough curves (comparison of Fig. I-1 with Fig. I-2). The solid curves in Fig. I-2 demonstrate that an increase in the feed flow rate (from 4.85 to 9.70 gpm/ft²) also increased the rate of breakthrough for all the nuclides involved. However, the relative effect of the flow rate on breakthrough behavior was not as pronounced as that of the salt concentration.

I.2 Breakthrough Behavior of $\text{Na}_2\text{HPO}_4\text{-H}_3\text{BO}_3\text{-LiOH}$ Type Feed

As shown in Fig. I-3, breakthrough curves for various ionic species in type 3 feed solution containing H_3BO_3 , LiOH , Na_2HPO_4 , ^{137}Cs , ^{131}I , and Sr are considerably different from those for types 1 and 2 feed solutions (see Table I-1) containing NaCl , ^{137}Cs , ^{131}I , and Sr . Although Na^+ , $^{137}\text{Cs}^+$, and $^{131}\text{I}^-$ ions in Fig. I-3 displayed the same type of breakthrough behavior as those in Figs. I-1 and I-2, the curves for Li^+ , BO_3^{3-} , and PO_4^{3-} reach points where the effluent concentrations exceed the feed concentrations. All three species go through individual maxima at different values of feed throughput.

In type 3 solution, the feed throughput at a Na^+ $\text{DF} = 10$ was about 92 b.v. (Fig. I-3) which compares with the value of 71 b.v. at the same DF for Na^+ in type 1 solution (Fig. I-1). The ratio of these two throughput volumes ($92/71 = 1.30$) is very close to the reciprocal of the concentration ratio of Na in the two solutions (394 ppm in type 1/324 ppm in type 3 = 1.22), indicating that the behavior of Na^+ is practically independent of Li^+ (~ 2 ppm) and the ^{137}Cs and Sr . It also shows that the feed throughput at the same DF for Na^+ is inversely proportional to the Na^+ concentration.

The ratios of effluent to feed concentrations greater than 1.0 which occur in the breakthrough curves for Li^+ , borates, and phosphates in the exhaustion or loading phase (see Figs. I-3, -4, and -5) might be partially explained by a chromatographic separation and concentration of these ions taking place in the ion exchange resin bed by virtue of differences in their distribution coefficients.

Although borates and phosphates in the effluent were calculated as BO_3^{3-} and PO_4^{3-} , there is evidence that they probably exist as mixtures of borates (e.g., H_2BO_3^- , HBO_3^{2-} , BO_3^{3-} , etc.) and phosphates (e.g., H_2PO_4^- , HPO_4^{2-} , PO_4^{3-} , etc.). Boric acid is only slightly ionized in solution at room temperature with an initial ionization constant for $\text{H}_3\text{BO}_3 \rightleftharpoons \text{H}^+ + \text{H}_2\text{BO}_3^-$ of 7.3×10^{-10} and constants for subsequent ionizations which are three to four orders of magnitude lower.

I.3 Relation Between Electrical Conductivity, pH, and Breakthrough

The variations in the specific electrical conductivity and pH of effluents during the exhaustion (loading) step are shown separately in Figs. I-6 and I-7 for the separate-bed and mixed-bed ion exchange systems. Dramatic changes in both conductivity and pH are seen to take place in the effluent from the anion unit of the separate-bed system (Fig. I-6). The two curves proceed in the opposite direction, forming almost a mirror image of each other. Thus, the conductivity underwent an upward change of more than two orders of magnitude while the pH shifted from 9.5 to 2.5. The changes in the conductivity and pH are much less drastic in the effluent from the cation unit.

A comparison of Fig. I-6 with Fig. I-1 shows that the peak of the conductivity curve for the anion column effluent nearly corresponds to the complete breakthrough of the Cl^- ion. Similarly, the complete breakthrough of the Na^+ ion approximately coincides with the minimum of the cation effluent conductivity curve. There appears to be no simple way of estimating intermediate DF values from either the conductivity curve or the pH curve unless a predetermined correlation is

available. Likewise neither curve reflects the extent of breakthrough of $^{137}\text{Cs}^+$ or $^{131}\text{I}^-$ because of the trace quantities involved.

In the mixed-bed system (Fig. I-7), a similar relation exists between the conductivity and pH. That is, the peak of the conductivity curve corresponds to the minimum point of the pH curve for column A. For column B, the minimum pH occurs at a feed throughput far beyond that of the conductivity peak. Upon comparison of Fig. I-7 with Figs. I-4 and I-5, it is seen that the maxima of the conductivity curves partially coincide with the maxima of the breakthrough curves for PO_4^{3-} with respect to the feed throughput.