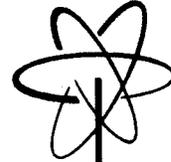


Miller

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MARCH 1973**



LICENSING TOPICAL

**TECHNICAL DERIVATION OF
BWR 1971 DESIGN BASIS
RADIOACTIVE MATERIAL
SOURCE TERMS**

**J. M. SKARPELOS
R. S. GILBERT**

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ABSTRACT

General Electric Company has evaluated radioactive material source terms in operating boiling water reactors (BWRs) over the past decade. These source terms are continually reviewed. The source terms used in design documents are periodically revised to incorporate new information.

This report reviews the 1971 radioactive material source terms and explains their derivation.

1. INTRODUCTION

General Electric Company has evaluated radioactive material source terms in operating boiling water reactors (BWRs) over the past decade. These source terms are continually reviewed. The source terms used in design documents are periodically revised to incorporate new information. Release of radioactive material from operating BWRs has generally resulted in doses to off-site persons which have been only a small fraction of permissible doses, or of natural background dose.

During 1970, Appendix I to regulation 10CFR50, which provides numerical guides to keep radioactivity in effluents as "low as practicable," and Appendix D, which implements the National Environmental Policy Act of 1969, was proposed. Concurrent with these proposed regulation revisions, numerous licensing questions were raised about radioisotopes and effluents previously considered negligible. To assist in answering these questions, the General Electric BWR radiation source terms were reviewed and updated in 1971. The 1971 source terms were made available in the public record on Docket 50-293.¹ Included are tabulations of noble gases, halogens, other fission products, and activation products. This reference also shows how the source terms are applied in describing the processing and release of effluents.

This report reviews the 1971 radioactive material source terms and explains their derivation. The radioisotope source terms discussed herein do not include coolant activation products such as N-13, N-16, N-17, O-19 and F-18, which are of significance to in-plant shielding but not to environmental releases.

The technical analyses and background information provided here are applicable to BWR/5 and BWR/6 plants.

2. BACKGROUND INFORMATION

The noble radiogas fission product leakage rates observed in operating BWRs are generally complex mixtures whose sources vary from miniscule defects in cladding to "tramp" uranium on external cladding surfaces. The distribution pattern or specific amounts of noble gas isotopes relative to the other noble gas isotopes can be described as follows:

$$\text{Equilibrium: } A_g \approx k_1 \gamma$$

$$\text{Recoil: } A_g \approx k_2 \gamma \lambda$$

The table of nomenclature at the end of this report defines the terms in these and succeeding equations. The constants k_1 and k_2 describe the fractions of the total fissions that are involved in each of the leakage rates. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that is physically possible. The equilibrium mixture results when a sufficient time delay exists between the fission event and the time of release of the noble radiogases from the fuel to the coolant for the radiogases to approach equilibrium levels in the fuel. When there is no delay or impedance between the fission event and the release of the radiogases, the recoil mixture is observed.

Prior to Vallecitos Boiling Water Reactor (VBWR) and Dresden 1 experience, it was assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel.

VBWR and early Dresden 1 experience indicated that the actual mixture most often observed approached a distribution which was intermediate in character to the two extremes. This intermediate decay mixture was termed the "diffusion" mixture. It must be emphasized that this "diffusion" mixture is merely one possible point on the mixture spectrum ranging from the equilibrium to the recoil mixture and does not have the absolute mathematical and mechanistic basis for the calculational methods possible for equilibrium and recoil mixtures. However, the "diffusion" distribution pattern has been described as follows:²

$$\text{Diffusion: } A_g \approx k_3 \gamma \lambda^{0.5}$$

The constant k_3 describes the fraction of total fissions that are involved in the release. As can be seen, the exponent of the decay constant, λ , is midway between that of equilibrium (0) and recoil (1). The "diffusion" pattern value of 0.5 was originally derived from diffusion theory, but the assumptions used are now doubtful.

3. NOBLE RADIOGAS FISSION PRODUCTS

3.1 NOBLE RADIOGAS DESIGN BASIS RELEASE MAGNITUDE

Although the previously described "diffusion" mixture was used as a basis for BWR design since 1963, the design basis release magnitude used has varied from 0.5 Ci/s to 0.1 Ci/s as measured after 30-minutes decay ($t = 30m$).* Since about 1967, the design basis release magnitude (including the 1971 source terms) has been established at an annual average of 0.1 Ci/s ($t = 30m$).

This design basis is considered as an annual average with some time above and some time below this value. This design value was selected on the basis of operating experience rather than predictive analysis. Several judgment factors including the significance of environmental release, reactor water radioisotope concentration, liquid waste handling and effluent disposal criteria, building air contamination, shielding design, and turbine and other component contamination affecting maintenance have been considered in establishing this level.

Experience in the operation of open cycle BWRs has indicated that some of these in-plant contamination and other operating restrictions may limit plant operation at levels well below emission rates which would correspond to 10CFR20 dose limits (500 mRem/y to any off-site person).

Although noble radiogas leakage rates above 0.1 Ci/s ($t = 30m$) can be tolerated for reasonable periods of time, long-term operation at such levels may be undesirable. Continual assessment of this value is made on the basis of actual operating experience in BWRs. When the 1971 design basis source terms were established, consideration was given to plant size and fuel power density and their possible affect on the design basis source terms. The conclusion at that time was that there was insufficient experimental or operational information available to provide a technical justification for changing the design basis release magnitude specifically because of plant size or fuel power density.

3.2 NOBLE RADIOGAS MIXTURE

While the noble radiogas design basis release magnitude is established at 0.1 Ci/s at 30-minutes decay, it is recognized that there may be a more statistically applicable distribution for the noble radiogas mixture. Sufficient data were available from KRB operations from 1967 to mid 1971 along with Dresden 2 data from operation in 1970 and for a short time in 1971 to more accurately characterize the noble radiogas mixture pattern for an operating BWR.

The initial equation used to analyze the collected leakage rate data is:

$$A_g = K_g \gamma \lambda^m (1 - e^{-\lambda T}) (e^{-\lambda t}). \quad (3-1)$$

With the exception of Kr-85 with a half-life of 10.74y, the noble radiogas fission products in the fuel approach an equilibrium condition after an irradiation period of several weeks (rate of formation is equal to the rate of decay). So for practical purposes, the term $(1 - e^{-\lambda T})$ approaches 1 and can be neglected when the reactor has been operating at steady state for long periods of time. The term $(e^{-\lambda t})$ is used to adjust the releases from the fuel ($t = 0$) to the decay time for which values are needed. Or conversely, the term is used to correct measured data at some known time to $t = 0$. Historically, $t = 30m$ has been used for design purposes. When discussing long steady-state operation and leakage rates from the fuel, the following simplified form of Equation 3-1 can be used to describe the leakage rate of each noble radiogas:

$$A_g = K_g \gamma \lambda^m. \quad (3-2)$$

*The noble radiogas source term rate after 30-minutes decay has been used as the conventional measure for describing the design basis release magnitude since it is conveniently measurable and was consistent with the nominal design basis 30-minute off-gas holdup system used on a number of plants. Over 90% of the total magnitude at 30-minutes decay can be generally accounted for by seven noble radiogas isotopes.

The constant, K_g , describes the magnitude of the leakage rate. The rate of noble radiogas leakage with respect to each other (composition) is expressed in terms of m , the exponent of the decay constant term, λ .

Multiplying both sides of Equation 3-2 by the term:

$$\left(\frac{3.7 \times 10^4}{\gamma\lambda}\right)$$

Handwritten notes: A_g $\frac{3.7 \times 10^4 \text{ atoms}}{\text{s}}$ $\left[\frac{\delta}{\lambda}\right] = R_g$ $\frac{\text{Sixtime}}{\text{s}}$

results in the following equation:

$$\frac{(3.7 \times 10^4)A_g}{\gamma\lambda} = (3.7 \times 10^4) K_g \lambda^{m-1}. \tag{3-3}$$

The microcurie is defined as that quantity of a radioactive nuclide disintegrating at the rate of 3.700×10^4 atoms per second. Therefore, the left-hand term in Equation 3-3 has the units of fissions/s, the source term, which is set equal to R_g . The value $(3.7 \times 10^4) K_g$ can be defined as K'_g and the value $(m - 1)$ is set equal to $(-b)$ to result in the following equation:

$$R_g = K'_g \lambda^{-b}. \tag{3-4}$$

Taking the logarithm of both sides of Equation 3-4 results in the following equation:

$$\log (R_g) = -b \log (\lambda) + \log (K'_g). \tag{3-5}$$

Equation 3-5 represents a straight line when $\log (R_g)$ is plotted versus $\log (\lambda)$; the term, $-b$, is the slope of the line and $\log (K'_g)$ is the value of R_g where the line intercepts the R_g axis at $\log (\lambda) = 0$. By plotting (R_g) versus (λ) on logarithmic graph paper, the same straight line is obtained. By fitting actual data obtained at KRB and Dresden 2 to Equation 3-4 (using least squares techniques), the slope $(-b)$ and K'_g are obtained. This technique is used to correlate the KRB and Dresden 2 data summarized in Tables 3-1 and 3-2.

With releases at KRB over the nearly 5-year period varying from 0.001 to 0.056 Ci/s ($t = 30m$) and with releases at Dresden 2 varying from 0.001 to 0.169 Ci/s ($t = 30m$) the average value of \bar{b} is determined. The value of b is 0.6 with a standard deviation of ± 0.07 . Since $(m-1) = -b$, the value of \bar{m} is 0.4 and the experimental data in terms of m is illustrated in Figure 3-1 as a frequency histogram. As can be seen from this figure, variations in m were observed in the range $m = 0.1$ to $m = 0.6$. After establishing the value for m , the value of K_g can be calculated by selecting a value for A_g , or as has been done historically, the design basis is set using the total design basis release magnitude at $t = 30m$. With ΣA_g at $30m = 100,000 \mu\text{Ci/s}$, K_g can be calculated as being 2.6×10^7 and Equation 3-1 becomes:

$$A_g = 2.6 \times 10^7 \gamma \lambda^{0.4} (1 - e^{-\lambda T}) (e^{-\lambda t}). \tag{3-6}$$

Using Equation 3-6, the resultant leakage rates are presented in Table 3-3 as released from fuel ($t = 0$) and after decay. This up-dated noble gas source term mixture has been termed the "1971 Mixture" to differentiate it from the "diffusion mixture".

3.3 LEAKAGE RATE OF Kr-85

Because of the long half-life of Kr-85 (10.74y), an equilibrium inventory is never reached. Because of its relatively low release, special measurement techniques are required and few attempts to measure Kr-85 have been made. The recombiner-charcoal off-gas treatment system at KRB provided an opportunity to measure this radionuclide. The leakage rate data for Kr-85 at several levels of total noble radiogas leakage is illustrated in Figure 3-2. Use of Equation 3-6 to calculate Kr-85 release, assuming a four-year fuel irradiation time, would tend to underestimate leakage rates when compared to observed KRB leakage rates. For that reason, a leakage rate range of 10 to 20 $\mu\text{Ci/s}$ was arbitrarily selected for design basis purposes.

Table 3-1a
SUMMARY OF KRB NOBLE RADIOGAS LEAKAGE RATES 1967*

Month	Power (MWt)	Sets of Data Included in Average	Average** Leakage Rate Before Delay Pipe ($\mu\text{Ci/s}$)	Average** Leakage Rate at $t = 30\text{m}$ ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b ***
January	—	—	—	—	—	—
February	800	1	2.4×10^{-3}	1.1×10^{-3}	8.83×10^8	0.854
March [†]	—	—	—	—	—	—
April	800	1	4.1×10^3	1.6×10^3	—	—
May	800	10	1.1×10^4	5.7×10^4	1.69×10^{10}	0.737
June	800	4	1.6×10^4	8.0×10^3	2.06×10^{10}	0.747
July	800	1	1.5×10^4	7.7×10^3	3.81×10^{10}	0.678
August [†]	—	—	—	—	—	—
September [†]	—	—	—	—	—	—
October	800	19	1.1×10^4	6.2×10^3	9.0×10^9	0.800
November	800	20	1.3×10^4	7.1×10^3	9.5×10^9	0.804
December	800	5	1.2×10^4	6.2×10^3	9.2×10^9	0.796

*Data obtained by KRB personnel under the direction of Dr. H. J. Schroeder.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

[†]Reactor down February 24 to March 25, and July 8 to September 25 (for fuel inspection).

Table 3-1b
SUMMARY OF KRB NOBLE RADIOGAS LEAKAGE RATES 1968*

Month	Power (MWt)	Sets of Data Included in Average	Average** Leakage Rate Before Delay Pipe ($\mu\text{Ci/s}$)	Average** Leakage Rate at $t = 30\text{m}$ ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b ***
January†	800	—	$\sim 1.2 \times 10^4$	$\sim 6.0 \times 10^3$	—	—
February†	800	—	$\sim 1.2 \times 10^4$	$\sim 6.0 \times 10^3$	—	—
March	800	3	1.7×10^4	9.0×10^3	1.20×10^{10}	0.817
	800	2	3.4×10^4	2.0×10^4	2.57×10^{10}	0.801
April	800	—	$\sim 3.6 \times 10^4$	$\sim 1.8 \times 10^4$	—	—
May†	—	—	—	—	—	—
June	770	4	3.6×10^4	2.1×10^4	9.62×10^{10}	0.685
July	770	4	3.8×10^4	2.1×10^4	5.79×10^{10}	0.736
August	770	4	3.2×10^4	1.6×10^4	5.82×10^{10}	0.723
September†	—	—	—	—	—	—
October†	—	—	—	—	—	—
November†	—	—	—	—	—	—
December	670	8	3.4×10^4	1.7×10^4	7.70×10^{10}	0.681

* Data obtained by KRB personnel under the direction of Dr. H. J. Schroeder.

** Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

*** Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

† Reactor down January 11 to February 15, May 9 to June 4, and August 13 to November 25 (for turbine repair).

Table 3-1c
SUMMARY OF KRB NOBLE RADIOGAS LEAKAGE RATES 1969*

Month	Power (MWt)	Sets of Data Included in Average	Average** Leakage Rate Before Delay Pipe ($\mu\text{Ci/s}$)	Average** Leakage Rate at t = 30m ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b***
January	680	7	4.2×10^4	2.2×10^4	1.26×10^{11}	0.657
February	680	6	4.0×10^4	2.1×10^4	7.80×10^{10}	0.705
March	680	3	6.0×10^4	3.1×10^4	1.87×10^{11}	0.654
	680	2	8.8×10^4	4.7×10^4	2.48×10^{11}	0.664
April	690	2	1.1×10^5	5.4×10^4	4.12×10^{11}	0.628
May	610-680	5	1.2×10^5	5.6×10^4	5.52×10^{11}	0.601
June [†]	—	—	—	—	—	—
July [†]	—	—	—	—	—	—
August [†]	—	—	—	—	—	—
September	780-790	6	3.5×10^4	1.3×10^4	4.56×10^{11}	0.462
October	780-790	6	3.2×10^4	1.3×10^4	2.20×10^{11}	0.540
November	780-790	4	3.4×10^4	1.5×10^4	1.39×10^{11}	0.600
December	780-790	6	5.0×10^4	2.4×10^4	1.41×10^{11}	0.653

*Data obtained by KRB personnel under the direction of Dr. H. J. Schroeder.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

[†]Reactor down for refueling May 29 to August 25.

Table 3-1d
SUMMARY OF KRB NOBLE RADIOGAS LEAKAGE RATES 1970*

Month	Power (MWt)	Sets of Data Included Average	Average** Leakage Rate Before Delay Pipe ($\mu\text{Ci/s}$)	Average** Leakage Rate at $t = 30\text{m}$ ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b ***
January	780-790	—	6.9×10^4	—	—	—
February	780-790	4	6.3×10^4	3.1×10^4	3.37×10^{11}	0.589
March	780-790	4	6.2×10^4	2.9×10^4	3.32×10^{11}	0.582
April	780-790	5	7.8×10^4	3.5×10^4	4.31×10^{11}	0.576
May	780-790	5	1.2×10^5	5.4×10^4	6.75×10^{11}	0.571
June†	—	—	—	—	—	—
July	780-790	3	4.1×10^4	1.5×10^4	7.03×10^{11}	0.416
August	780-790	4	3.7×10^4	1.5×10^4	2.98×10^{11}	0.523
September	780-790	4	4.7×10^4	2.1×10^4	1.58×10^{11}	0.626
October	780-790	5	4.6×10^4	2.1×10^4	1.25×10^{11}	0.647
November	780-790	4	4.5×10^4	2.1×10^4	1.24×10^{11}	0.652
December	780-790	5	5.2×10^4	2.6×10^4	8.63×10^{10}	0.714

* Data obtained by KRB personnel under the direction of Dr. H. J. Schroeder.

** Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

*** Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

† Reactor down for refueling May 29 to July 25.

Table 3-1e
SUMMARY OF KRB NOBLE RADIOGAS LEAKAGE RATES 1971*

	Power (MWt)	Sets of Data Included in Average	Average** Leakage Rate Before Delay Pipe ($\mu\text{Ci/s}$)	Average** Leakage Rate at t = 30m ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b***
January	780-790	4	4.9×10^4	2.4×10^4	1.28×10^{11}	0.663
February	780-790	4	5.9×10^4	3.1×10^4	1.44×10^{11}	0.680
March	780-790	4	5.9×10^4	3.1×10^4	1.67×10^{11}	0.662
April	780-790	5	7.6×10^4	4.0×10^4	1.83×10^{11}	0.678
May	780-790	4	8.1×10^4	4.1×10^4	2.80×10^{11}	0.637
June†	—	—	—	—	—	—
July†	—	—	—	—	—	—
August†	—	—	—	—	—	—
September†	—	—	—	—	—	—
October†	—	—	—	—	—	—
November†	—	—	—	—	—	—
December†	—	—	—	—	—	—

*Data obtained by KRB personnel under the direction of Dr. H. J. Schroeder.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

†Reactor down May 30 for refueling, last part of year data not used in establishing 1971 source terms.

Table 3-2a
SUMMARY OF DRESDEN 2 NOBLE RADIOGAS LEAKAGE RATES*

Date	Power (MWt)	Leakage Rate** at SJAE ($\mu\text{Ci/s}$)	Leakage Rate** at t = 30m ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b***
Apr. 29, 1970	1210	1.0×10^4	4.8×10^3	1.11×10^{11}	0.502
May 3, 1970	920	7.0×10^3	3.3×10^3	5.56×10^{11}	0.552
May 3, 1970	920	7.0×10^3	—	5.71×10^{10}	0.547
May 7, 1970	742	6.9×10^3	—	2.14×10^{10}	0.643
May 17, 1970	1247	2.6×10^4	1.3×10^4	1.17×10^{11}	0.609
May 18, 1970	1835	7.5×10^4	4.0×10^4	3.17×10^{11}	0.623
May 19, 1970	1857	1.1×10^5	—	2.63×10^{11}	0.687
May 20, 1970	1950	1.6×10^5	7.6×10^4	3.25×10^{12}	0.446
May 27, 1970	1628	9.1×10^4	—	8.62×10^{11}	0.522
May 27, 1970	1846	1.3×10^5	—	9.86×10^{11}	0.553
May 28, 1970	1768	8.1×10^4	—	1.35×10^{11}	0.708
Jun. 2, 1970	1865	3.4×10^4	—	5.44×10^{11}	0.473
Jun. 4, 1970	1698	3.1×10^4	1.5×10^4	3.05×10^{11}	0.525

*Data obtained by CECO and GE personnel.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

Table 3-2b
SUMMARY OF DRESDEN 2 NOBLE RADIOGAS LEAKAGE RATES*

Date	Power (MWt)	Leakage Rate** at SJAE ($\mu\text{Ci/s}$)	Leakage Rate** at t = 30m ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b***
Aug. 13, 1970	637	2.2×10^3	9.0×10^2	1.25×10^{11}	0.309
Aug. 15, 1970	1300	5.0×10^3	2.1×10^3	1.02×10^{11}	0.431
Aug. 17, 1970	1285	4.5×10^3	1.9×10^3	4.26×10^{10}	0.516
Aug. 21, 1970	1293	8.0×10^3	3.6×10^3	3.06×10^{10}	0.604
Aug. 24, 1970	1730	2.7×10^4	1.2×10^4	1.07×10^{11}	0.608
Aug. 24, 1970	1730	2.6×10^4	1.2×10^4	8.23×10^{10}	0.633
Aug. 26, 1970	1852	4.6×10^4	2.1×10^4	1.03×10^{11}	0.669
Aug. 28, 1970	1854	3.6×10^4	1.8×10^4	6.18×10^{10}	0.713
Aug. 30, 1970	529	6.5×10^3	3.5×10^3	2.35×10^9	0.848
Aug. 31, 1970	1214	9.0×10^3	3.9×10^3	4.18×10^{10}	0.593

*Data obtained by CECO and GE personnel.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

Table 3-2c
 SUMMARY OF DRESDEN 2 NOBLE RADIOGAS LEAKAGE RATES*

Date	Power (MWt)	Leakage Rate** at SJAE ($\mu\text{Ci/s}$)	Leakage Rate** at t = 30m ($\mu\text{Ci/s}$)	K'_g *** (fissions)	b***
Sept. 1, 1970	1563	1.6×10^4	7.0×10^3	8.12×10^{10}	0.585
Sept. 2, 1970	1576	1.7×10^4	8.0×10^3	1.00×10^{11}	0.571
Sept. 3, 1970	1594	1.9×10^4	8.0×10^3	8.39×10^{10}	0.591
Sept. 4, 1970	1579	1.6×10^4	7.0×10^3	8.07×10^{10}	0.581
Sept. 8, 1970	1849	3.7×10^4	1.7×10^4	1.42×10^{11}	0.613
Sept. 9, 1970	1610	2.9×10^4	1.4×10^4	3.64×10^{10}	0.730
Sept. 10, 1970	1705	2.7×10^4	1.2×10^4	1.33×10^{11}	0.582
Sept. 11, 1970	1633	4.6×10^4	2.2×10^4	8.86×10^{10}	0.688
Sept. 14, 1970	1878	5.1×10^4	2.1×10^4	3.36×10^{11}	0.533
Sept. 21, 1970	1770	4.9×10^4	2.1×10^4	1.31×10^{11}	0.637
Sept. 23, 1970	2140	1.2×10^5	5.6×10^4	2.19×10^{11}	0.689
Oct. 7, 1970	2170	1.2×10^5	5.4×10^4	9.45×10^{11}	0.537
Oct. 9, 1970	2350	2.6×10^5	1.2×10^5	8.83×10^{11}	0.625
Oct. 12, 1970	2400	1.9×10^5	8.3×10^4	4.73×10^{11}	0.646
Oct. 13, 1970	2483	2.6×10^5	1.2×10^5	7.53×10^{11}	0.646

*Data obtained by CECO and GE personnel.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit to equation $R_g = K'_g \lambda^{-b}$ by R. S. Gilbert.

Table 3-2d
SUMMARY OF DRESDEN 2 NOBLE RADIOGAS LEAKAGE RATES*

Date	Power (MWt)	Leakage Rate** at SJAE ($\mu\text{Ci/s}$)	Leakage Rate** at t = 30m ($\mu\text{Ci/s}$)	Kg*** (fissions)	b***
Nov. 4, 1970	1218	2.7×10^4	1.1×10^4	5.68×10^{11}	0.415
Nov. 12, 1970	1164	1.2×10^4	5.0×10^3	8.24×10^{10}	0.525
Nov. 13, 1970	1176	1.9×10^4	7.0×10^3	6.15×10^{10}	0.566
Nov. 26, 1970	1872	1.4×10^5	6.8×10^4	7.30×10^{11}	0.586
Dec. 3, 1970	1804	8.0×10^4	3.9×10^4	1.32×10^{11}	0.707
Dec. 29, 1970	1741	8.9×10^4	4.1×10^4	1.14×10^{12}	0.479
Jan. 6, 1971	1756	6.1×10^4	2.9×10^4	2.87×10^{11}	0.597
Jan. 7, 1971	1762	8.9×10^4	4.3×10^4	3.42×10^{11}	0.619
Jan. 8, 1971	1830	8.3×10^4	3.8×10^4	3.06×10^{11}	0.620
Jan. 13, 1971	1824	1.1×10^5	4.9×10^4	5.99×10^{11}	0.570
Jan. 14, 1971	1852	2.7×10^5	1.2×10^5	1.41×10^{12}	0.562

*Date obtained by CECO and GE personnel.

** Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

*** Least squares fit to equation $R_g = K_g' \lambda^{-b}$ by R. S. Gilbert.

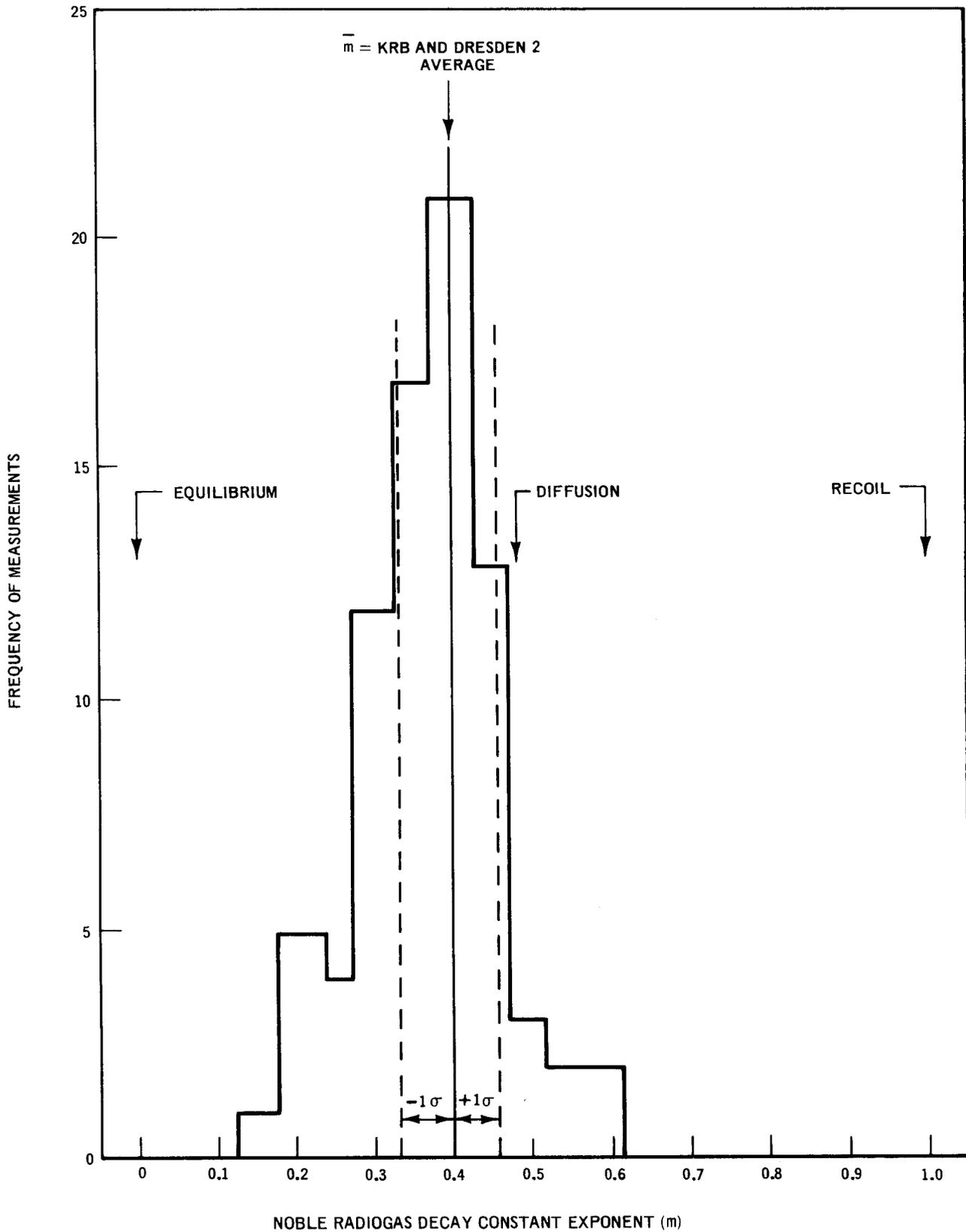


Figure 3-1. Noble Radiogas Decay Constant Exponent Frequency Histogram

Table 3-3
NOBLE RADIOGAS LEAKAGE RATES FOR ALL BWR/5 AND BWR/6 PLANTS

Radioisotope	Half-life*	Decay Constant λ (s^{-1})	Yield** U_{th}^{235} (atoms/fission)	Leakage Rate at $t = 0$ ($\mu Ci/s$)	Leakage Rate at $t = 30 m$ ($\mu Ci/s$)
Kr-83m	1.86 h	1.03×10^{-4}	5.2×10^{-3}	3.4×10^3	2.9×10^3
Kr-85m	4.4 h	4.38×10^{-5}	1.3×10^{-2}	6.1×10^3	5.6×10^3
Kr-85	10.74 y	2.05×10^{-9}	2.71×10^{-3}	10 to 20 †	10 to 20 †
Kr-87	76 m	1.52×10^{-4}	2.53×10^{-2}	2.0×10^4	1.5×10^4
Kr-88	2.79 h	6.90×10^{-5}	3.56×10^{-2}	2.0×10^4	1.8×10^4
Kr-89	3.18 m	3.63×10^{-3}	4.59×10^{-2}	1.3×10^5	1.8×10^2
Kr-90	32.3 s	2.15×10^{-2}	5.0×10^{-2}	2.8×10^5	—
Kr-91	8.6 s	8.06×10^{-2}	3.45×10^{-2}	3.3×10^5	—
Kr-92	1.84 s	3.77×10^{-1}	1.87×10^{-2}	3.3×10^5	—
Kr-93	1.29 s	5.37×10^{-1}	4.8×10^{-3}	9.3×10^4	—
Kr-94	1.0 s	6.93×10^{-1}	1.0×10^{-3}	2.3×10^4	—
Kr-95	0.5 s	1.39	7.0×10^{-5}	2.1×10^3	—
Kr-97	1 s	6.93×10^{-1}	$<6 \times 10^{-7}$	1.4×10^1	—
Xe-131m	11.96 d	6.71×10^{-7}	1.7×10^{-4}	1.5×10^1	1.5×10^1
Xe-133m	2.26 d	3.55×10^{-6}	1.6×10^{-3}	2.9×10^2	2.8×10^2
Xe-133	5.27 d	1.52×10^{-6}	6.69×10^{-2}	8.2×10^3	8.2×10^3
Xe-135m	15.7 m	7.36×10^{-4}	1.8×10^{-2}	2.6×10^4	6.9×10^3
Xe-135	9.16 h	2.10×10^{-5}	6.3×10^{-2}	2.2×10^4	2.2×10^4
Xe-137	3.82 m	3.02×10^{-3}	6.0×10^{-2}	1.5×10^5	6.7×10^2
Xe-138	14.2 m	8.13×10^{-4}	5.9×10^{-2}	8.9×10^4	2.1×10^4
Xe-139	40 s	1.73×10^{-2}	5.4×10^{-2}	2.8×10^5	—
Xe-140	13.6 s	4.90×10^{-2}	3.8×10^{-2}	3.0×10^5	—
Xe-141	1.72 s	4.03×10^{-1}	1.33×10^{-2}	2.4×10^5	—
Xe-142	1.22 s	5.68×10^{-1}	3.5×10^{-3}	7.3×10^4	—
Xe-143	.96 s	7.22×10^{-1}	5.1×10^{-4}	1.2×10^4	—
Xe-144	9 s	7.7×10^{-2}	6.0×10^{-5}	5.6×10^2	—
TOTALS				$\sim 2.5 \times 10^6$	$\sim 1.0 \times 10^5$

*Half-life taken from *Chart of the Nuclides – Tenth Edition* by F. W. Walker and N. E. Holden, 1969.

**Yield taken from *Summary of Fission Project Yields for U-235, Pu-239, and Pu-241 at Thermal, Fission Spectrum and 14 MeV Neutron Energies*, by M. E. Meek and B. F. Rider, APED-5398-A, October 1, 1968.

†Estimated from experimental observations.

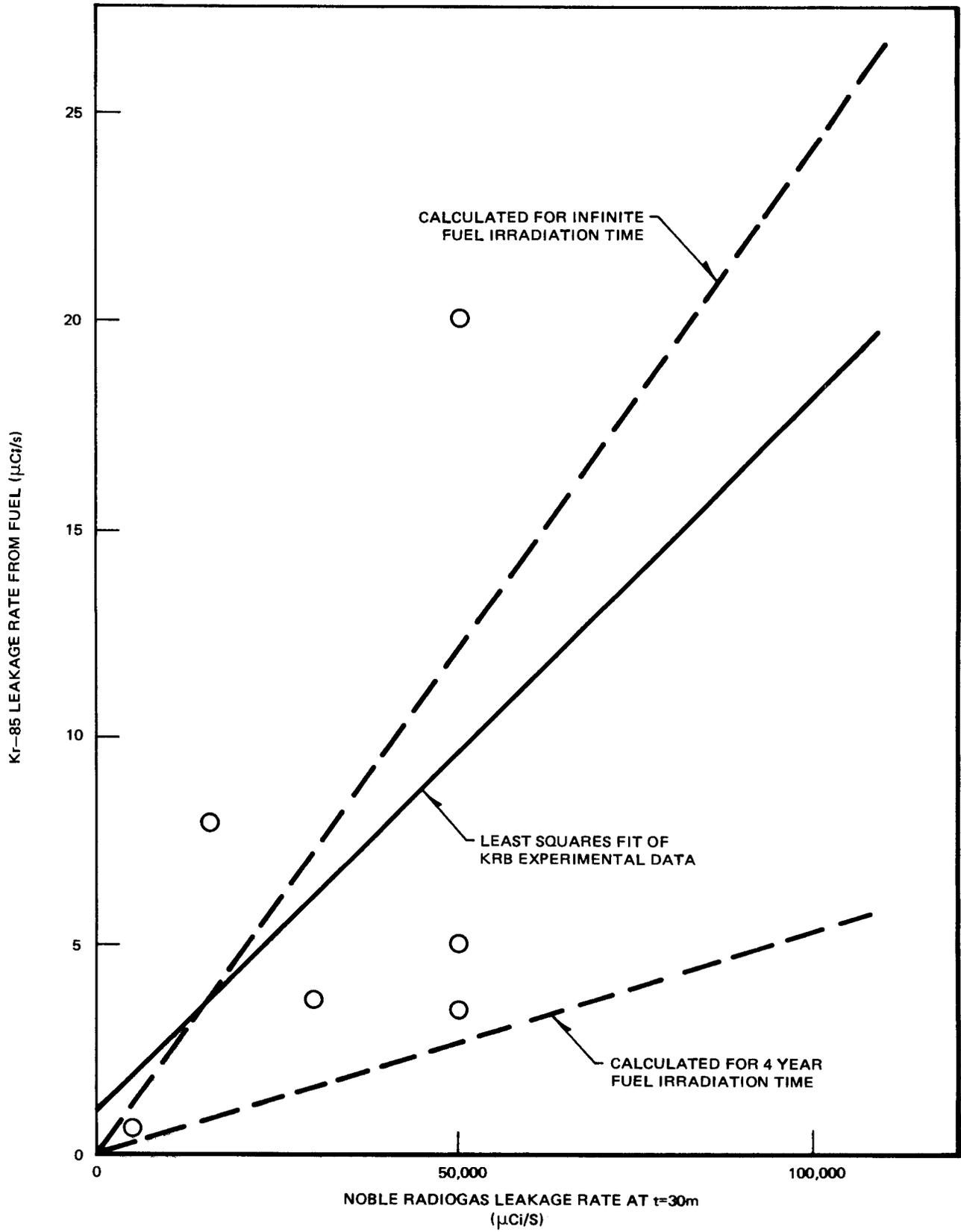


Figure 3-2. Kr-85 Leakage Rate vs Noble Radiogas Leakage Rate

4. RADIOHALOGEN FISSION PRODUCTS

4.1 RADIOHALOGEN MIXTURE

Historically, the radiohalogen design basis leakage rates for each halogen radioisotope were established using the same equation as that used for calculating noble radiogas leakage rates. In a fashion similar to that used with radiogases, a simplified equation can be shown to describe the leakage rate of each halogen radioisotope:

$$A_h = K_h \gamma \lambda^n. \quad (4-1)$$

The constant, K_h , describes the magnitude of leakage rate from fuel. The rate of radiohalogen leakage with respect to each other (composition) is expressed in terms of n , the exponent of the decay constant, λ .

Using a mathematically analogous method as is used for noble radiogases, the following equation is derived:

$$R_h = K'_h \gamma \lambda^{-b}. \quad (4-2)$$

In this case $(n - 1)$ is set equal to $-b$. The actual data obtained from KRB and Dresden 2 is applied to Equation 4-2 (using least squares techniques) to obtain the slope $(-b)$ and K'_h . This technique is used to correlate the KRB and Dresden 2 data summarized in Tables 4-1 and 4-2.

As was done with the noble radiogases, the average value of b is determined. The value of \bar{b} for radiohalogens is 0.5 with a standard deviation of ± 0.19 . Since $(n - 1) = -b$, the value of \bar{n} is also 0.5. The experimental data in terms of \bar{n} is illustrated in Figure 4-1 as a frequency histogram. As can be seen from this figure, variations in n were observed in the range of $n = 0.1$ to $n = 0.9$.

4.2 RADIOHALOGEN DESIGN BASIS RELEASE MAGNITUDE

As mentioned above, it appeared that the use of the previous method of calculating radiohalogen leakage from fuel was overly conservative. Figure 4-2 relates KRB and Dresden 2 noble radiogas leakage rate versus I-131 leakage rate. While it can be seen from Dresden 2 data during the period August 1970 to January 1971 that there is a relationship between noble radiogas leakage rates and I-131 leakage rates under one fuel condition, there is no simple relationship for all fuel conditions experienced. Also, it can be seen that during this period, high radiogas leakage rates were not accompanied by high radioiodine leakage rates from the fuel. Except for one KRB datum point, all steady-state I-131 leakage rates observed at KRB or Dresden 2 were equal to or less than $505 \mu\text{Ci/s}$. Even at Dresden 1 in March 1965, when severe defects were experienced in stainless-steel-clad fuel, I-131 leakage rates greater than $500 \mu\text{Ci/s}$ were not experienced. Figure 4-2 shows that these higher radioiodine leakages from the fuel were related to noble radiogas leakage rates of less than the design basis value of 0.1 Ci/s ($t = 30\text{m}$). This may be explained by inherent limitations due to internal plant operational problems that caused plant derating. In general, one would not anticipate continued operation at full power for any significant time period with fuel cladding defects which would be indicated by I-131 leakage rates from the fuel in excess of $700 \mu\text{Ci/s}$. When high radiohalogen leakage rates are observed, other fission products will be present in greater amounts. This may increase potential radiation exposure to operating and maintenance personnel during plant outages following such operation.

Using these judgment factors and experience to date, the design basis radiohalogen source term leakage rate from fuel was established based on I-131 leakage of $700 \mu\text{Ci/s}$. This value, as seen in Figure 4-2, accommodates the experience data and the design basis noble radiogas source term of 0.1 Ci/s ($t = 30\text{m}$). With the I-131 design basis source term established, K_h is calculated as being 2.4×10^7 and halogen radioisotopes leakage rates are expressed by the following equation:

$$A_h = 2.4 \times 10^7 \gamma \lambda^{0.5} (1 - e^{-\lambda T})(e^{-\lambda t}). \quad (4-3)$$

Using Equation 4-3, the resultant leakage rates are presented in Table 4-3 as released from fuel ($t = 0$).

Table 4-1
SUMMARY OF KRB RADIOHALOGEN LEAKAGE RATES*

Year	Month	Average** Noble Radiogas Leakage Rate at t = 30 min ($\mu\text{Ci/s}$)	I-131 Leakage Rate at t = 0 ($\mu\text{Ci/s}$)	b***
1967	February	1.1×10^3	9.0	0.76
	April	1.6×10^3	2.1×10^1	—
	May	5.7×10^3	2.2×10^1	0.82
	July	7.7×10^3	1.7×10^2	—
	September	6.0×10^3	2.2×10^1	—
	November	7.1×10^3	3.1×10^1	0.52
1968	March	9.0×10^3	3.6×10^1	0.54
	April	1.8×10^4	4.7×10^1	—
	July	2.1×10^4	4.4×10^1	0.63
	August	1.6×10^4	1.1×10^2	—
	December	1.7×10^4	1.1×10^2	—
1969	March	3.1×10^4	4.6×10^2	0.71
	March	4.7×10^4	1.7×10^3	0.62
	September	1.3×10^4	3.6×10^1	0.22
	October	1.3×10^4	3.8×10^1	0.27
	November	1.5×10^4	5.5×10^1	0.29
	December	2.4×10^4	2.0×10^2	0.44
1970	January	3.5×10^4	2.2×10^2	—
	February	3.1×10^4	2.4×10^2	0.43
	March	2.9×10^4	2.1×10^2	0.41
	April	3.5×10^4	2.8×10^2	0.39
	May	5.4×10^4	3.7×10^2	0.42
	August	1.5×10^4	9.0×10^1	0.32
	September	2.1×10^4	9.9×10^1	0.33
	October	2.1×10^4	9.2×10^1	0.35
	November	2.1×10^4	9.6×10^1	0.34
	December	2.6×10^4	9.5×10^1	0.34
1971	January	2.4×10^4	1.1×10^2	0.34
	February	3.1×10^4	1.8×10^2	0.42
	April	4.0×10^4	1.9×10^2	0.43

*Data obtained by KRB personnel under the direction of Dr. H. J. Schroeder.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit of five radioisotopes (I-131, I-132, I-133, I-134, I-135) to the equation $R_h = K'_h \lambda^{-b}$.

Table 4-2
SUMMARY OF DRESDEN 2 RADIOHALOGEN LEAKAGE RATES*

Date	Average** Noble Radiogas Leakage Rate at t = 30 min ($\mu\text{Ci/s}$)	I-131 Leakage Rate at t = 0 ($\mu\text{Ci/s}$)	b***
April 29, 1970	4.8×10^3	3.7×10^1	0.54
May 3, 1970	3.3×10^3	7.8×10^{-1}	0.16
May 17, 1970	1.3×10^4	7.1×10^1	0.62
May 18, 1970	4.0×10^4	2.3×10^2	0.75
May 20, 1970	7.6×10^4	3.0×10^2	0.87
May 27, 1970	5.6×10^4	5.1×10^2	0.92
June 4, 1970	1.5×10^4	1.5×10^2	0.89
Aug 13, 1970	9.0×10^2	5.3×10^{-1}	0.09
Aug 15, 1970	2.1×10^3	1.7	0.34
Aug 17, 1970	1.9×10^3	3.8	0.47
Aug 26, 1970	2.1×10^4	5.8×10^1	0.72
Sept. 1, 1970	7.0×10^3	3.4×10^1	0.67
Sept. 2, 1970	8.0×10^3	4.7×10^1	0.54
Sept. 23, 1970	5.6×10^4	8.0×10^1	0.73
Oct. 7, 1970	5.4×10^4	6.1×10^1	0.56
Oct. 9, 1970	1.2×10^5	1.0×10^2	0.65
Oct. 12, 1970	8.3×10^4	1.1×10^2	0.53
Nov. 4, 1970	1.1×10^4	1.5×10^1	0.34
Nov. 12, 1970	5.0×10^3	1.6×10^1	0.57
Jan. 26, 1971	3.6×10^4	6.1×10^1	0.47
Jan. 29, 1971	1.4×10^5	1.4×10^2	0.57

*Data obtained by Commonwealth Edison and GE personnel directed by Dr. J. H. Holloway.

**Summation of seven radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135m, Xe-135, Xe-138).

***Least squares fit of five radioisotopes (I-131, I-132, I-133, I-134, I-135) to the equation

$$R_h = K_h \lambda^{-b}.$$

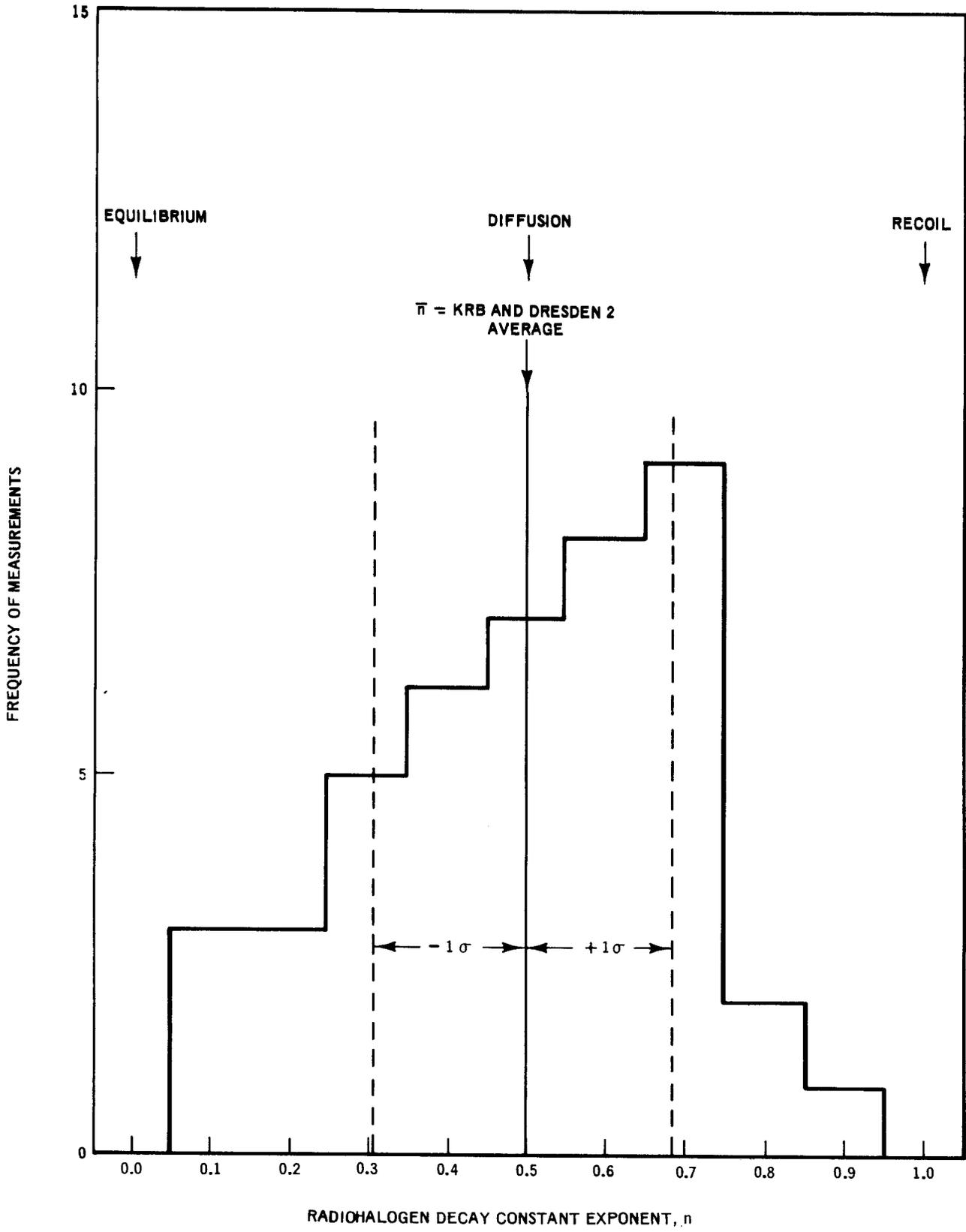


Figure 4-1. Radiohalogen Decay Constant Exponent Frequency Histogram

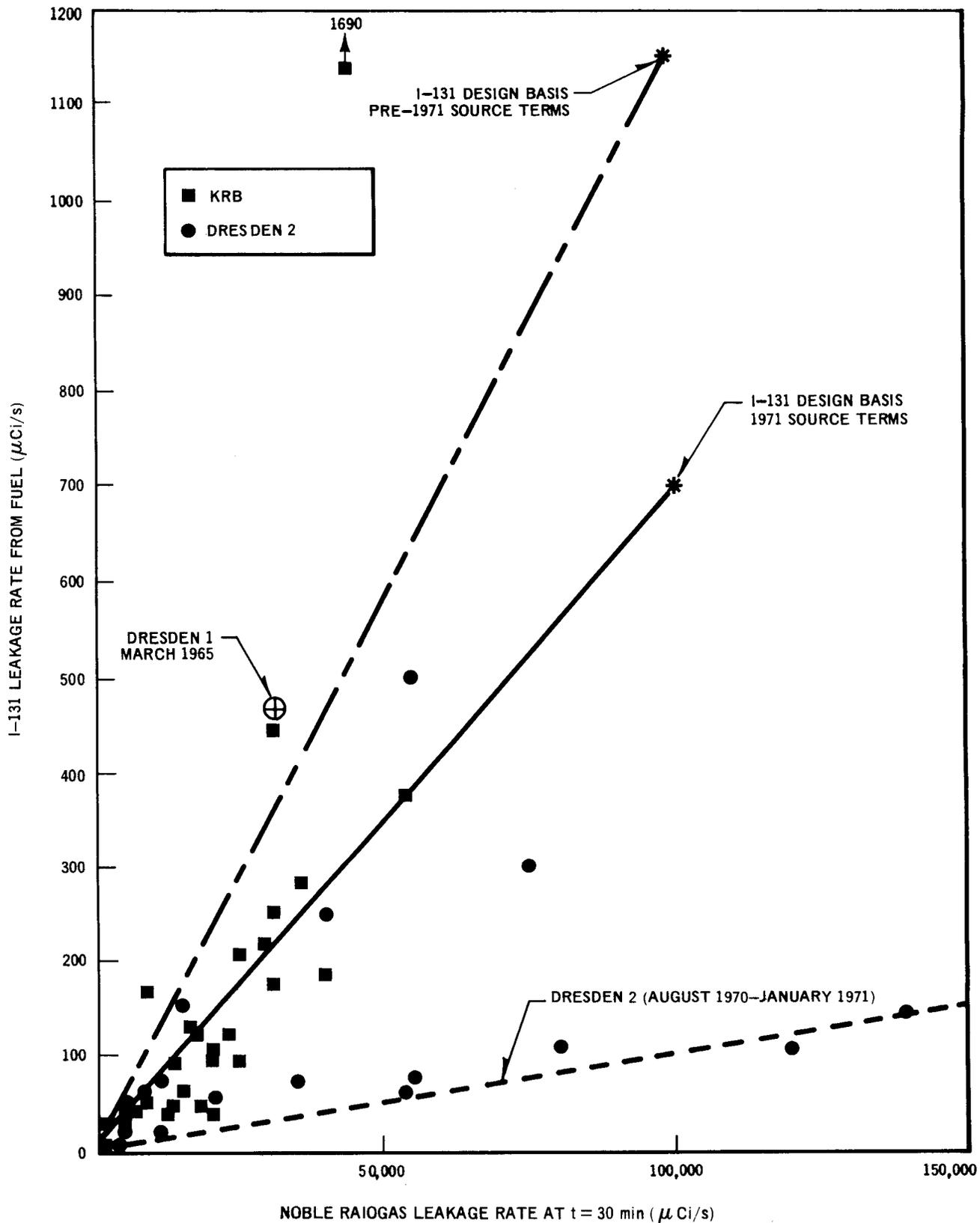


Figure 4-2. Noble Radiogas Leakage Rates vs I-131 Leakage Rate

Table 4-3
BWR/5 STANDARD PLANTS DESIGN BASIS RADIOHALOGEN LEAKAGE RATES

Radioisotope	Half-Life*	Decay Constant λ (s^{-1})	Yield** U^{235}_{th} (atoms/fission)	Leakage Rate at $t = 0$ ($\mu Ci/s$)
Br-83	2.40 h	8.02×10^{-5}	5.2×10^{-3}	1.1×10^3
Br-84	31.8 m	3.63×10^{-4}	9.3×10^{-3}	4.3×10^3
Br-85	3.0 m	3.85×10^{-3}	1.30×10^{-2}	1.9×10^4
I-131	8.065 d	9.95×10^{-7}	2.91×10^{-2}	7.0×10^2
I-132	2.284 h	8.43×10^{-5}	4.26×10^{-2}	9.4×10^3
I-133	20.8 h	9.26×10^{-6}	6.69×10^{-2}	4.9×10^3
I-134	52.3 m	2.21×10^{-4}	7.8×10^{-2}	2.8×10^4
I-135	6.7 h	2.87×10^{-5}	6.17×10^{-2}	7.9×10^3

*Half-life taken from *Chart of the Nuclides – Tenth Edition* by F. W. Walker and N. E. Holden, 1969.

**Yield taken from *Summary of Fission Product Yields for U-235, U-238, Pu-239 and Pu-241 at Thermal, Fission Spectrum and 14 MeV Neutron Energies*, by M. E. Meek and B. F. Rider, APED-5398-A, October 1, 1968.

4.3 RADIOHALOGEN REACTOR WATER CONCENTRATIONS

Bromine radioisotopes leakage rates were established using the same equation although only one measurement was available to verify this choice. Concentrations of halogens in reactor water can be calculated using the following equation:

$$C_h = \frac{A_h}{M\lambda + \epsilon f + \phi F} \tag{4-3}$$

Using Equation 4-4 and leakage rates shown in Table 4-3 along with plant parameters shown in Table 4-4, the design basis radiohalogen concentrations for three BWR/5 standard plants are calculated. These are shown in Table 4-5.

Equation 4-3 provides the long-term steady-state equilibrium concentration (not to be confused with the equilibrium mixture previously mentioned). Experience with mixed deep bed demineralizers indicates that the removal efficiency for radiohalogens is greater than 0.9 (90%). For design purposes, ϵ was set equal to 1 since the uncertainty in the carryover fraction, ϕ , would influence the concentration more significantly.

4.4 RADIOIODINE CARRYOVER

The carryover, ϕ , for a particular species is defined as the ratio of its concentration in a condensed steam sample or a hotwell condensate sample to its concentration in the reactor coolant. This technique has been used for many years to determine steam quality using Na-24 as the tracer. Carryover fraction based upon this radioisotope has generally been measured at $\phi < 0.001$ (<0.1%) on plants having both internal and external steam separators and dryers. At KRB where internal steam separation and drying was first applied, higher radioiodine carryover fractions relative to Na-24 carryover fractions were observed. On later plants, during the startup phase, carryover fractions up to 0.01 were observed. Up to early 1971, the average carryover fraction for all available data from Oyster Creek and later plants indicated 0.012 ± 0.009 . More recent evaluations suggest the average carryover fraction to be 0.018 ± 0.004 . Table 4-6 summarizes I-131 carryover measurements.

**Table 4-4
PLANT PARAMETERS USED IN THIS STUDY**

Parameter	Units	KRB	Dresden 2	Standard Plant BWR/5	Standard Plant BWR/5	Standard Plant BWR/5
Plant size	*	—	251-724	201-444	218-560	251-764
Rated power	MWt	801	2527	1931	2436	3323
Steam Flow, F	kg/s	290	1235	1046	1320	1801
Cleanup Flow, f	kg/s	10	37.8	10.5	12.6	16.8
Reactor Water Inventory, M	kg	1.00×10^5	2.63×10^5	1.723×10^5	2.008×10^5	2.666×10^5
Carryover Fraction, ϕ						
For Halogens	—	0.01	0.02	0.02	0.02	0.02
For other radioisotopes	—	<0.001	<0.001	<0.001	<0.001	<0.001

*Plant size = Nominal reactor vessel diameter in inches — Number of fuel bundles.

Table 4-5
BWR/5 STANDARD PLANTS
CALCULATED DESIGN BASIS RADIOHALOGEN CONCENTRATIONS ($\mu\text{Ci}/\text{kg}$)

Plant Size	201-444	218-560	251-764
Rated Power (MWt)	1931	2436	3323
Radioisotope:			
Br-83	2.4×10^1	2.0×10^1	1.5×10^1
Br-84	4.6×10^1	3.8×10^1	2.9×10^1
Br-85	2.7×10^1	2.3×10^1	1.8×10^1
I-131	2.2×10^1	1.8×10^1	1.3×10^1
I-132	2.1×10^2	1.7×10^2	1.3×10^2
I-133	1.5×10^2	1.2×10^2	8.9×10^1
I-134	4.0×10^2	3.4×10^2	2.5×10^2
I-135	2.2×10^2	1.8×10^2	1.3×10^2

Table 4-6
SUMMARY OF RADIOIODINE CARRYOVER

Plant	Carryover Fraction, ϕ^*	
	Startup Data	Mature Plant Data
Nine Mile Point	0.01	0.02**
Dresden 2	0.016	0.018
Dresden 3	0.018	—
Oyster Creek	0.006	0.018**
Tsuruga †	0.008 to 0.022**	—
Millstone	0.011	0.017
Nuclenor	0.002	—
Monticello	0.005	—

* Ratio of I-131 in condensate to that measured in reactor coolant unless otherwise noted.

** High values confirmed heater drain samples and "no cleanup" test.

† Ratio of I-131 in steam sample so that measured in reactor coolant.

5. NONVOLATILE SOLUBLE FISSION PRODUCTS

When the 1971 design basis source terms were derived, the radioisotopes of Sr, Mo, Tc, Te, Cs, Ba, and Np were studied as a group. It was believed at that time that essentially all of the leakage rate for the radioisotopes of these elements could be accounted for by the recoil mechanisms from fuel present in the coolant or deposited on fuel and component surfaces. Certain radioisotopes are produced in the coolant by the decay of their precursor gas as well as by direct release from the fuel adding to the complexity of deriving the rate of release of these nuclides from their observed concentrations. Leakage rate of a recoil distribution is characterized by the following equation:

$$A_r = K_r \gamma \lambda; \quad (5-1)$$

or the source term can be described as follows:

$$R_r = \frac{3.7 \times 10^4 (A_r)}{\gamma \lambda} = (3.7 \times 10^4) (K_r) = K_r'. \quad (5-2)$$

Evidence for the existence of a recoil component has been demonstrated by the following observations:

- a. Turbine contamination by Ce-141 measured at certain plants can only be explained by assuming its source to be the short-lived precursor Xe-141.
- b. Analysis of steam and condensate for the decay products of other short-lived precursors also indicate a high release component for these gases.
- c. Data from Humboldt and Dresden 1 in 1965 indicate that the leakage rate of all the soluble fission products measured with half-lives less than 10.5d can be described by Equation 5-1 and their source term by Equation 5-2.
- d. Measurement of noble radiogas distribution during initial operation following extended outages show significant Xe-133 leakage rates relative to the remaining gases.

An important observation of a recoil source term is that the leakage rates vary linearly with reactor power levels, in contrast to the delayed (inventory controlled) source terms which increase exponentially with power. Although the recoil mechanism must apply equally to all fission product nuclides, the latter are not observed to a significant extent in sampled reactor coolant.

There appears to be no consistent relationship between a recoil leakage rate of noble radiogases and observed summation of noble radiogases measured at 30m decay. Table 5-1 presents the recoil source term observed at several plants, the calculated noble radiogas leakage rates that this recoil source would produce and the observed noble radiogas leakage rate. Since the highest recoil source term can be explained by severe fuel failure when stainless steel cladding was used, the selection of a recoil source term of 3×10^{13} fissions/s conservatively describes the release of nonvolatile soluble fission products.

This technique could not adequately describe all the nonvolatile soluble fission product levels observed, so the source terms for these radioisotopes measured in some 18 KRB samples shown in Table 5-2 were compared with the radiohalogen source term. The correlation is shown in Figure 5-1 with source terms ratioed and normalized to a common I-131 source term of $700 \mu\text{Ci/s}$. A few measurements for these radioisotopes made at Dresden 2 are listed in Table 5-3; however, these data were not included in the correlation. The final equation used to define the design basis non-volatile soluble nuclide source term was the sum of one-tenth of that used to define radiohalogens plus the recoil component:

$$R_s = 0.1 R_h + R_r, \quad (5-3)$$

or

$$A_s = 0.1 A_h + A_r. \quad (5-4)$$

Table 5-1
SUMMARY OF OBSERVED RECOIL SOURCE TERMS

Plant	Time Period	Recoil Source Term R_r (fissions/s)	Noble Radiogas Leakage Rate*	
			Calculated From Recoil (μ Ci/s)	Actual Observed (μ Ci/s)
Humboldt	August 1967	4.0×10^{13}	2.4×10^4	2.7×10^4
	August 1968	4.0×10^{13}	2.4×10^4	4.0×10^4
Dresden 1	March 1965	1.0×10^{14}	6.0×10^4	7.0×10^4
	January 1968	1.0×10^{13}	6.0×10^3	1.0×10^4
	February 1969	1.0×10^{13}	6.0×10^3	2.7×10^4
	June 1969	1.0×10^{13}	6.0×10^3	4.4×10^4
	August 1969	1.0×10^{13}	6.0×10^3	5.1×10^4
KRB	Feb.—June 1967	3.0×10^{11}	2.0×10^2	$1-15.0 \times 10^3$
	November 1967	2.0×10^{12}	1.2×10^3	—
	Mar.—Aug. 1968	3.0×10^{12}	2.0×10^3	—
	March 1969	2.0×10^{13}	1.2×10^4	—
	Mar.—Dec. 1969	2.0×10^{13}	1.2×10^4	—
	Jan.—Mar. 1970	2.0×10^{13}	1.2×10^4	$1.3-5.0 \times 10^4$
	May 1970	4.0×10^{13}	2.4×10^4	—
	July 1970	2.0×10^{13}	1.2×10^4	—
August 1970	2.0×10^{13}	1.2×10^4	—	
Dresden 2	January 1971	1.0×10^{13}	6.0×10^3	1.0×10^6
	May 1971	6.0×10^{12}	4.0×10^3	—
Oyster Creek	January 1970	5.0×10^{11}	3.0×10^2	4.0×10^3
Big Rock	March 1966	5.0×10^{13}	3.1×10^4	3.3×10^4
	September 1966	6.0×10^{13}	3.7×10^4	4.6×10^4
	Mar.—Jun. 1968	3.0×10^{12}	2.0×10^3	$5-15.0 \times 10^3$

*Summation of six radioisotopes (Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, Xe-138) Measured at $t = 30$ min.

Table 5-2a
 RADIOHALOGEN AND NONVOLATILE SOLUBLE FISSION
 PRODUCT CONCENTRATION SUMMARY FOR KRB 1966-67 ($\mu\text{Ci/kg}$)

Date	Dec. 2, 1966	Feb. 1, 1967	Feb. 23, 1967	April 3, 1967	July 5, 1967	Sept. 28, 1967	Nov. 14 1967
Reactor Power (MWt)	500	800	800	800	800	800	800
Halogens:							
Br-84	—	—	—	—	—	—	—
I-131	6.0×10^{-4}	7.1×10^{-1}	8.1×10^{-1}	1.6	1.3×10^1	1.7	2.4
I-132	—	2.1	2.5	1.9	9.2	9.9	1.0×10^1
I-133	5.0×10^{-3}	3.7	1.1	4.3	2.6×10^1	1.3×10^1	1.1×10^1
I-134	3.3×10^{-2}	3.0	1.2	6.9	2.3×10^1	3.2×10^1	3.9×10^1
I-135	6.0×10^{-3}	3.8	1.3	2.6	3.4×10^1	1.4×10^1	1.2×10^1
Nonvolatile Soluble Fission Products:							
Sr-89	—	—	—	—	—	—	—
Sr-90	—	—	—	—	—	—	—
Sr-91	1.3×10^{-2}	1.1	9.1×10^{-1}	4.3×10^{-1}	4.2	3.4	3.8
Sr-92	2.9×10^{-2}	3.0	8.1×10^{-2}	5.4×10^{-1}	8.2	9.3	1.3×10^1
Mo-99	5.0×10^{-2}	3.0×10^{-1}	2.2×10^{-1}	3.4×10^{-1}	1.7	1.1	1.9
Tc-99m	—	—	—	—	—	—	4.5
Tc-101	—	—	—	—	—	—	5.1×10^1
Te-129m	—	—	—	—	—	—	—
Te-132	—	—	—	—	—	—	3.0×10^{-2}
Cs-134	—	—	—	—	—	—	—
Cs-136	—	—	—	—	—	—	1.6×10^{-2}
Cs-137	—	—	—	—	—	—	1.2×10^{-2}
Cs-138	—	—	—	—	—	—	—
Ba-139	2.0×10^{-1}	2.0×10^{-1}	3.5×10^{-1}	—	6.1	1.0×10^1	6.5
Ba-140	—	1.5×10^{-2}	—	3.4×10^{-2}	2.5×10^{-1}	3.1×10^{-1}	5.8×10^{-1}
Ba-141	—	—	—	—	—	—	—
Ba-142	—	—	—	—	—	—	—
Np-239	5.0×10^{-2}	2.0	2.7	2.9	8.3	1.1×10^1	1.6×10^1

Table 5-2b
 RADIOHALOGEN AND NONVOLATILE SOLUBLE FISSION
 PRODUCT CONCENTRATION SUMMARY FOR KRB 1968 ($\mu\text{Ci}/\text{kg}$)

Date	Mar 12, 1968	Apr 25, 1968	July 8, 1968	Aug 12, 1968	Dec 18, 1968
Reactor Power (MWt)	800	800	770	770	670
Halogens:					
Br-84	—	—	—	—	—
I-131	2.8	3.6	3.4	8.8	8.6
I-132	8.0	1.5×10^1	9.9	2.2×10^1	3.4×10^1
I-133	1.0×10^1	1.4×10^1	9.4	2.6×10^1	2.5×10^1
I-134	1.6×10^1	3.3×10^1	—	5.2×10^1	7.8×10^1
I-135	1.8×10^1	1.9×10^1	1.3×10^1	2.9×10^1	4.4×10^1
Nonvolatile Soluble Fission Products:					
Sr-89	—	—	—	7.2×10^{-1}	6.9×10^{-1}
Sr-90	—	—	—	8.2×10^{-2}	1.1×10^{-1}
Sr-91	4.0	4.3	4.4	1.4	7.9
Sr-92	6.6	9.8	1.0×10^1	2.1×10^1	2.0×10^1
Mo-99	2.5	2.5	2.5	5.1	4.4
Tc-99 m	7.4	1.2×10^1	6.2	1.1×10^1	1.5×10^1
Tc-101	5.7×10^1	9.7×10^1	—	—	1.9×10^2
Te-129 m	—	—	—	—	—
Te-132	4.0×10^{-2}	9.2×10^{-2}	—	—	$<1.0 \times 10^{-2}$
Cs-134	6.7×10^{-3}	4.2×10^{-2}	4.5×10^{-2}	7.0×10^{-2}	9.3×10^{-2}
Cs-136	5.1×10^{-3}	2.9×10^{-2}	—	—	6.6×10^{-2}
Cs-137	1.2×10^{-2}	6.4×10^{-2}	7.3×10^{-2}	7.8×10^{-2}	1.4×10^{-1}
Cs-138	—	—	—	—	3.1
Ba-139	1.1×10^1	1.7×10^1	1.2×10^1	1.5×10^1	1.8×10^1
Ba-140	2.7×10^{-1}	3.7×10^{-1}	3.5×10^{-1}	9.0×10^{-1}	9.2×10^{-1}
Ba-141	1.6×10^1	1.2×10^2	3.5×10^1	7.4×10^1	6.4×10^1
Ba-142	1.9×10^1	2.3×10^2	6.9×10^1	7.9×10^1	6.8×10^1
Np-239	1.1×10^1	2.2×10^1	3.6×10^1	7.4	2.1×10^1

Table 5-2c
 RADIOHALOGEN AND NONVOLATILE SOLUBLE FISSION
 PRODUCT CONCENTRATION SUMMARY FOR KRB 1969-70-71 ($\mu\text{Ci/kg}$)

Date	Mar 8, 1969	Mar 20, 1969	Sept 20, 1969	Jan 16, 1970	Apr 21, 1970	Feb 10, 1971
Reactor Power (MWt)	670	670	800	780	780	790
Halogens:						
Br-84	—	—	—	3.1×10^1	—	—
I-131	3.5×10^1	1.3×10^2	3.0	1.7×10^1	2.1×10^1	5.1
I-132	1.4×10^2	4.4×10^2	8.4×10^1	2.0×10^2	3.3×10^2	6.4×10^1
I-133	1.3×10^2	3.8×10^2	3.3×10^1	1.1×10^2	1.2×10^2	3.1×10^1
I-134	—	8.3×10^2	2.4×10^2	4.0×10^2	6.4×10^2	1.9×10^2
I-135	1.2×10^2	5.1×10^2	9.5×10^1	3.5×10^2	2.9×10^2	8.5×10^1
Nonvolatile Soluble Fission Products:						
Sr-89	—	3.5	—	—	—	4.0×10^{-1}
Sr-90	—	4.2×10^{-1}	—	—	—	1.0×10^{-2}
Sr-91	1.9×10^1	3.3×10^1	2.6×10^1	1.4×10^1	3.2×10^1	3.5
Sr-92	4.8×10^1	5.8×10^1	5.0×10^1	2.9×10^1	6.2×10^1	1.5×10^1
Mo-99	3.4×10^1	5.5×10^1	8.0	2.6×10^1	3.8×10^1	8.0
Tc-99 m	5.4×10^1	2.1×10^2	3.8×10^1	1.3×10^2	2.7×10^2	6.2×10^1
Tc-101	—	1.2×10^3	5.8×10^1	9.2×10^2	1.2×10^3	6.6×10^2
Te-129 m	—	—	—	—	—	—
Te-132	—	5.8	1.3	—	—	—
Cs-134	—	5.4×10^{-1}	2.8×10^{-2}	2.4×10^{-1}	1.9×10^{-1}	1.1×10^{-1}
Cs-136	—	3.2×10^{-1}	2.6×10^{-2}	1.4×10^{-1}	1.8×10^{-1}	6.6×10^{-2}
Cs-137	—	9.0×10^{-1}	3.6×10^{-2}	2.7×10^{-1}	3.0×10^{-1}	1.5×10^{-1}
Cs-138	—	1.1×10^2	1.2×10^1	1.3×10^1	1.6×10^1	1.8×10^1
Ba-139	5.4×10^1	7.9×10^1	6.0×10^1	4.7×10^1	8.7×10^1	9.6×10^1
Ba-140	1.7	2.6	1.6	2.8	6.0	3.0
Ba-141	—	2.0×10^2	3.1×10^2	2.5×10^2	4.0×10^2	1.6×10^2
Ba-142	—	3.7×10^2	2.8×10^2	2.4×10^2	9.2×10^1	3.5×10^{-1}
Np-239	5.0×10^1	8.4×10^1	1.5×10^2	6.0×10^1	2.7×10^2	2.1×10^1

Table 5-3
 RADIOHALOGEN AND NONVOLATILE SOLUBLE FISSION
 PRODUCT CONCENTRATION SUMMARY FOR DRESDEN 2 1970-71 ($\mu\text{Ci}/\text{kg}$)

Date	Oct 12, 1970	Jan 26, 1971	Jan 29, 1971	Jun 30, 1971	Jun 26 1971
Reactor Power (MWt)	2400	1834	2211	2000	2000
Halogens:					
Br-84	—	—	—	—	—
I-131	1.7	1.1	2.3	2.5	1.5
I-132	8.5	7.5	1.2×10^1	1.1×10^1	—
I-133	5.6	4.8	8.6	7.0	—
I-134	3.5×10^1	3.0×10^1	3.4×10^1	1.3×10^1	—
I-135	7.6	1.1×10^1	1.4×10^1	8.4	—
Nonvolatile soluble Fission Products:					
Sr-89	—	3.3×10^{-2}	4.6×10^{-2}	—	—
Sr-90	—	2.1×10^{-3}	3.3×10^{-3}	—	—
Sr-91	1.4	3.4	3.2	1.2	—
Sr-92	7.0	1.1×10^1	1.0×10^1	5.1	—
Mo-99	—	1.6	—	—	1.0
Tc-99m	—	6.3	—	—	—
Tc-101	—	4.2×10^1	—	1.2×10^1	—
Te-129m	—	—	—	—	—
Te-132	—	—	—	—	—
Cs-134	—	3.0×10^{-4}	5.0×10^{-4}	—	2.1×10^{-3}
Cs-136	—	—	—	—	3.5×10^{-3}
Cs-137	—	3.0×10^{-3}	5.0×10^{-3}	—	1.2×10^{-3}
Cs-138	—	4.5	2.5	—	—
Ba-139	2.4	7.5	6.8	3.5	—
Ba-140	1.6×10^{-1}	2.3×10^{-1}	3.1×10^{-1}	—	1.2×10^{-1}
Ba-141	—	9.2	6.7	2.4	2.0×10^1
Ba-142	—	—	—	—	—
Np-239	7.8	1.6×10^1	1.4×10^1	3.7	3.6

The leakage rate is calculated from equation:

$$A_s = (2.4 \times 10^6 \gamma \lambda^{0.5}) + (8.1 \times 10^8 \gamma \lambda) \quad (5-5)$$

The leakage rates calculated using this equation are listed in Table 5-4.

Three radioisotopes, Cs-134, Cs-136 and Np-239 could not be accommodated directly with this equation since they are not direct fission products. Actually, they can be said to be activation products of stable cesium isotopes resulting from fission in the case of the cesium radioisotopes or activation of uranium in the case of neptunium. Arbitrarily, Cs-134 and Cs-136 were ratioed to Cs-137 on the basis of the four samples having the highest Cs-137 leakage rates. A fictitious yield of 0.6 atoms/fission was assigned to Np-239 which fit the curve selected for the other radioisotopes. This value can be explained as being the yield corresponding to the capture to fission ratio. This ratio increases from about 0.5 in a new core to about 1.3 in a core of 20,000 MWd/T exposure, and varies considerably between the surface and interior of the fuel itself.

The concentration in reactor coolant is calculated from the following equation:

$$C_s = \frac{A_s}{M\lambda + f} \quad (5-6)$$

and results are shown in Table 5-5 for several BWR/5 standard plants. Carryover with steam for these radioisotopes is assumed to be negligible and is not used in the calculation.

Carryover of nonvolatile soluble fission product radioisotopes from the reactor coolant to the steam is estimated to be negligible (<0.001 fraction). In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor will result in production of noble radiogas daughter radioisotopes in the steam and condensate systems. Certain short-lived rubidium and cesium radioisotopes have been omitted from listing, although their presence is obvious, because they are not normally measured and are of minor concern with plant effluents.

Some daughter isotopes (e.g., yttrium and lanthanum) were not listed as being in reactor water. Their independent release to the coolant is negligible; however, the isotope may be observed in some samples in the amounts expected from decay of the parent isotope.

Except for Np-239, trace concentrations of transuranic isotopes have been observed in only a few samples where extensive and complex analyses were carried out. The alpha activity present in reactor water is predominantly from Cm-242 at an estimated concentration of 10^{-6} Ci/g or less, which is below the maximum permissible concentration in drinking water. Alpha activity from plutonium isotopes is more than one order of magnitude lower than the activity from Cm-242. Plutonium-241 (primarily a beta emitter) may also be present in concentrations comparable to the Cm-242 level.

Table 5-4
 BWR/5 STANDARD PLANTS
 CALCULATED NONVOLATILE SOLUBLE FISSION PRODUCT LEAKAGE RATES

Radioisotope	Half-life*	Decay Constant λ (s^{-1})	Yield** U_{th}^{235} (atoms/fission)	Leakage Rate at $t = 0$ (μ Ci/s)
Sr-89	50.8 d	1.58×10^{-7}	4.76×10^{-2}	5.2×10^1
Sr-90	28.9 y	7.60×10^{-10}	5.83×10^{-2}	3.9
Sr-91	9.67 h	1.99×10^{-5}	5.81×10^{-2}	1.6×10^3
Sr-92	2.69 h	7.16×10^{-5}	5.3×10^{-2}	4.2×10^3
Mo-99	66.6 h	2.89×10^{-6}	6.16×10^{-2}	4.0×10^2
Tc-99m	6.007 h	3.21×10^{-5}	5.4×10^{-2}	2.1×10^3
Tc-101	14.2 m	8.14×10^{-4}	5.02×10^{-2}	3.7×10^4
Te-129 m	34.1 d	3.35×10^{-7}	3.5×10^{-3}	5.8
Te-132	78 h	2.47×10^{-6}	4.26×10^{-2}	2.5×10^2
Cs-134	2.06 y	1.07×10^{-8}	***	2.7
Cs-136	13 d	6.17×10^{-7}	***	1.8
Cs-137	30.2 y	7.28×10^{-10}	6.2×10^{-2}	4.1
Cs-138	32.2 m	3.59×10^{-4}	6.7×10^{-2}	2.3×10^4
Ba-139	83.2 m	1.39×10^{-4}	6.48×10^{-2}	9.1×10^3
Ba-140	12.8 d	6.27×10^{-7}	6.3×10^{-2}	1.5×10^2
Ba-141	18.3 m	6.31×10^{-4}	6.1×10^{-2}	3.5×10^4
Ba-142	10.7 m	1.08×10^{-3}	5.79×10^{-2}	5.5×10^4
Np-239	2.35 d	3.41×10^{-6}	6.0×10^{-1} ***	4.3×10^3

*Half-life taken from *Chart of the Nuclides—Tenth Edition* by F. W. Walker and N. E. Holden, 1969.

**Yield taken from *Summary of Fission Product Yields for U-235, U-238, Pu-239 and Pu-241 at Thermal, Fission Spectrum and 14 MeV Neutron Energies*, by M. E. Meek and B. F. Rider, APED-5398-A, October 1, 1968.

***See text for these special cases.

Table 5-5
BWR/5 STANDARD PLANTS
CALCULATED NONVOLATILE SOLUBLE FISSION PRODUCT CONCENTRATIONS ($\mu\text{Ci}/\text{kg}$)

Plant Size	201-444	218-560	251-764
Rated Power (MWt)	1931	2436	3323
Radioisotope:			
Sr-89	4.9	4.1	3.1
Sr-90	3.7×10^{-1}	3.1×10^{-1}	2.3×10^{-1}
Sr-91	1.1×10^2	9.1×10^1	6.9×10^1
Sr-92	1.8×10^2	1.4×10^2	1.1×10^2
Mo-99	3.6×10^1	3.0×10^1	2.2×10^1
Tc-99m	1.3×10^2	1.1×10^2	8.4×10^1
Tc-101	2.4×10^2	2.1×10^2	1.6×10^2
Te-129m	5.5×10^{-1}	4.6×10^{-1}	3.4×10^{-1}
Te-132	2.3×10^1	1.9×10^1	1.4×10^1
Cs-134	2.6×10^{-1}	2.1×10^{-1}	1.6×10^{-1}
Cs-136	1.7×10^{-1}	1.4×10^{-1}	1.1×10^{-1}
Cs-137	3.9×10^{-1}	3.2×10^{-1}	2.4×10^{-1}
Cs-138	3.2×10^2	2.4×10^2	1.9×10^2
Ba-139	2.6×10^2	2.1×10^2	1.6×10^2
Ba-140	1.4×10^1	1.2×10^1	9.0
Ba-141	2.9×10^2	2.2×10^2	1.7×10^2
Ba-142	2.8×10^2	2.1×10^2	1.7×10^2
Np-239	3.9×10^2	3.2×10^2	2.4×10^2

6. INSOLUBLE FISSION PRODUCTS

The estimation of the apparent source term for the group of radionuclides termed insoluble fission products is related to the most commonly measured (and most readily observed) radioisotope Zr-95. The insoluble fission product source terms is calculated using the following equation:

$$R_z = C_z(M\lambda + f) \left[\frac{3.7 \times 10^4}{y\lambda} \right]. \quad (6-1)$$

Thirty-three measurements for the Zr-95 source term at five BWR plants are summarized in Table 6-1. The logarithmic average for all observations was 3.3×10^{12} fissions/s. Other insoluble fission product source terms were calculated in a similar fashion and their source term value relative to Zr-95 is illustrated in Figure 6-1. From this figure, it can be seen that the insoluble fission products are distributed about an equilibrium pattern (i.e., $R_z \approx ky$), the same as would be observed if these radionuclides were present in microscopic particles of fuel. The amount of "fuel" can be related to the fission source term previously calculated. Similarly the psuedo leakage rates for these insoluble fission products are calculated using the relationship:

$$A_z = \frac{R_z y \lambda}{3.7 \times 10^4} \quad (6-2)$$

or more simply

$$A_z = 10.9y. \quad (6-3)$$

This equation assumes irradiation times long compared to the half-life involved. The specific psuedo leakage rates calculated for BWR analyses are shown in Table 6-2 and resultant concentrations calculated from the following equation:

$$C_z = \frac{A_z}{M\lambda + f} \quad (6-4)$$

are shown in Table 6-3 for several BWR/5 standard plants. Carryover with steam for these radioisotopes is assumed to be negligible and is not used in the calculation.

Table 6-1
SUMMARY OF Zr-95 SOURCE TERM OBSERVATIONS
AT SEVERAL PLANTS

Plant	Number of Samples	R_z (fissions/s)
Dresden 1	14	8.5×10^{12}
Dresden 2	2	1.3×10^{13}
KRB	15	8.4×10^{11}
Big Rock	2	8.2×10^{13}
Humboldt	2	4.2×10^{12}
All Plants	33	3.3×10^{12}

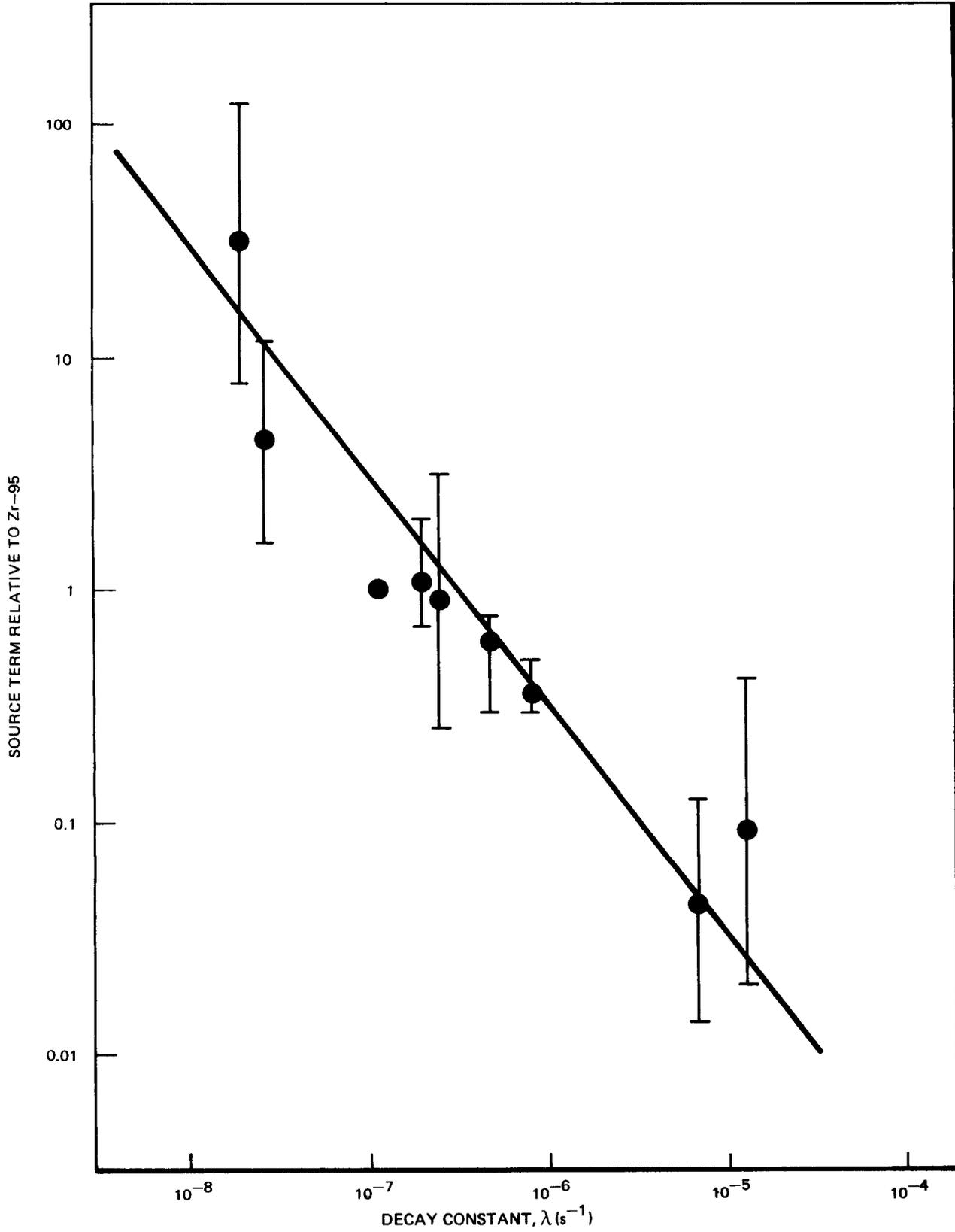


Figure 6-1. Insoluble Fission Product Source Term Relative to Zr-95 vs the Decay Constant

Table 6-2
BWR/5 STANDARD PLANTS CALCULATED INSOLUBLE FISSION PRODUCT LEAKAGE RATES

Radioisotope	Half-Life*	Decay Constant λ (s^{-1})	Yield** U^{235}_{th} (atoms/fission)	Pseudo Leakage Rate at $t = 0$ ($\mu Ci/s$)
Zr-95	65.5 d	1.23×10^{-7}	6.2×10^{-2}	6.8×10^{-1}
Zr-97	16.8 h	1.15×10^{-5}	5.9×10^{-2}	6.5×10^{-1}
Nb-95	35.1 d	2.29×10^{-7}	6.41×10^{-2}	7.0×10^{-1}
Ru-103	39.8 d	2.02×10^{-7}	3.0×10^{-2}	3.3×10^{-1}
Ru-106	368 d	2.18×10^{-8}	3.99×10^{-3}	4.4×10^{-2}
Ce-141	32.53 d	2.47×10^{-7}	6.1×10^{-2}	6.7×10^{-1}
Ce-143	33.0 h	5.83×10^{-6}	5.91×10^{-2}	6.5×10^{-1}
Cd-144	284.4 d	2.82×10^{-8}	5.40×10^{-2}	5.9×10^{-1}
Pr-143	13.58 d	5.91×10^{-7}	5.91×10^{-2}	6.5×10^{-1}
Nd-147	11.06 d	7.25×10^{-7}	2.19×10^{-2}	2.4×10^{-1}

*Half-life taken from *Chart of the Nuclides – Tenth Edition* by F. W. Walker and N. E. Holden, 1969.

**Yield taken from *Summary of Fission Product Yields for U-235, U-238, Pu-239 and Pu-241 at Thermal, Fission Spectrum and 14 MeV Neutron Energies*, by M. E. Meek and B. F. Rider, APED-5398-A, October 1, 1968.

Table 6-3
BWR/5 STANDARD PLANTS CALCULATED INSOLUBLE FISSION PRODUCT CONCENTRATIONS ($\mu Ci/kg$)

Plant Size	201-444	218-560	251-764
	1931	2436	3323
Radioisotope:			
Zr-95	6.5×10^{-2}	5.4×10^{-2}	4.0×10^{-2}
Zr-97	5.2×10^{-2}	4.2×10^{-2}	3.2×10^{-2}
Nb-95	6.6×10^{-2}	5.5×10^{-2}	4.2×10^{-2}
Ru-103	3.1×10^{-2}	2.6×10^{-2}	1.9×10^{-2}
Ru-106	4.2×10^{-3}	3.5×10^{-3}	2.6×10^{-3}
Ce-141	6.4×10^{-2}	5.3×10^{-2}	3.9×10^{-2}
Ce-143	5.6×10^{-2}	4.6×10^{-2}	3.5×10^{-2}
Ce-144	5.6×10^{-2}	4.7×10^{-2}	3.5×10^{-2}
Pr-143	6.1×10^{-2}	5.1×10^{-2}	3.8×10^{-2}
Nd-147	2.3×10^{-2}	1.9×10^{-2}	1.4×10^{-2}

7. ACTIVATION PRODUCTS

As indicated in Section 1, the coolant activation products are not treated in this report. Activation products formed by activation of impurities in the coolant or by corrosion of irradiated system materials are formed in a complex manner qualitatively understood, but not adequately correlated by simple equations. Table 7-1 lists the design basis concentrations which were established in the mid 1960's based on early Dresden 1 experience.³ The USPHS results⁴ obtained in 1968 at Dresden are also included. Additional data from KRB and Dresden 2, available early in 1971, were not sufficient to modify the design basis concentrations which, in general, are considered conservative.

Table 7-1
SUMMARY OF NONCOOLANT ACTIVATION PRODUCT CONCENTRATIONS (μ Ci/kg)

Radioisotope	Half-life	Design Basis All Plants	Dresden 1			
			GE 1963	USPHS 1968	KRB 1967-1971	Dresden 2 1970
Na-24	15.00h	2	2	1.6 – 3.0	–	–
P-32	14.31d	2×10^{-2}	2×10^{-2}	–	–	–
Cr-51	27.8d	5×10^{-1}	5×10^{-1}	$\sim 5 \times 10^{-1}$	1.1×10^{-1}	4.3
Mn-54	313d	4×10^{-2}	–	1.6×10^{-3}	1.8×10^{-2}	6.0×10^{-2}
Mn-56	2.582h	5×10^1	5×10^1	–	2.1	2.0×10^{-1}
Fe-55	2.7y	–	–	0.040 to 0.095	–	–
Fe-59	45d	8×10^{-2}	8×10^{-2}	–	3.3×10^{-2}	3.0×10^{-2}
Co-57	271d	–	–	$\sim 1 \times 10^{-2}$	$\sim 4 \times 10^{-4}$	–
Co-58	71.4d	5	5	1.7 to 14	1.5×10^{-1}	1.7×10^{-1}
Co-60	5.258y	5×10^{-1}	5×10^{-1}	0.26 to 2.2	2.6×10^{-2}	6.3×10^{-2}
Cu-64	12.75h	–	6×10^1	$\sim 1 \times 10^{-2}$	2.8×10^1	–
Ni-65	243.7d	3×10^{-1}	$< 3 \times 10^{-1}$	–	$< 1 \times 10^{-2}$	–
Zn-65	2.55h	2×10^{-3}	$< 2 \times 10^{-3}$	4.0×10^{-3}	1.3×10^{-2}	4×10^{-3}
Zn-69m	13.7h	3×10^{-2}	$< 3 \times 10^{-2}$	–	–	–
Ag-110m	253d	6×10^{-2}	6×10^{-2}	1.9×10^{-3}	4×10^{-3}	–
Ta-182	115d	–	–	2×10^{-2}	8×10^{-3}	–
W-187	23.9h	3	3	–	3×10^{-2}	–

8. TRITIUM

In a BWR, tritium is produced by three principal methods:

- a. activation of naturally occurring deuterium in the primary coolant;
- b. nuclear fission of UO_2 fuel; and
- c. neutron reactions with boron used in reactivity control rods.

With regard to tritium, which may be released from a BWR in liquid or gaseous effluents, the tritium formed in control rods which is released is believed to be negligible. A prime source of tritium available for release from a BWR is that produced from activation of deuterium in the primary coolant. Some fission product tritium may also transfer from fuel to primary coolant. This discussion is limited to the uncertainties associated with estimating the amounts of tritium generated in a BWR which are available for release.

All of the tritium produced by activation of deuterium in the primary coolant is available for release in liquid or gaseous effluents. The tritium formed in a BWR can be calculated using the equation:

$$A_{\text{at}} = \frac{\Sigma\Phi V\lambda}{3.7 \times 10^4 P} \quad (8-1)$$

For recent BWR designs, A_{at} is calculated to be $1.3 \pm 0.4 \times 10^{-4} \mu\text{Ci/s/MWt}$. The uncertainty indicated is derived from the estimated errors in selecting values for the coolant volume in the core, coolant density in the core, abundance of deuterium in light water (some additional deuterium will be present because of the $\text{H}(\eta, \gamma) \text{D}$ reaction), thermal neutron flux, and microscopic cross section for deuterium.

The fraction of tritium produced by fission which may transfer from fuel to the coolant (which will then be available for release in liquid and gaseous effluents) is much more difficult to estimate. However, since zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium will remain in the fuel rods⁵ unless defects are present in the cladding material.

The study made at Dresden 1 in 1968 by the U.S. Public Health Service⁴ suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium activation source. For purposes of estimating the leakage of tritium from defected fuel, we can make the assumption that it leaks in a manner similar to the leakage of noble radiogases. We can thus use the empirical relationship described as the "diffusion mixture" used for predicting the source term of individual noble gas radioisotopes as a function of total noble gas source term. The equation which describes this relationship is:

$$A_{\text{dt}} = K_t y \lambda^{0.5} \quad (8-2)$$

If the total noble radiogas source term is $10^5 \mu\text{Ci/s}$ after 30-m decay, we would calculate leakage of tritium from fuel to be about $0.24 \mu\text{Ci/s}$. To place this value in perspective in the USPHS study, the observed rate of Kr-85 (which has a half-life similar to that of tritium) was 0.06 to 0.4 times that calculated using the "diffusion mixture" relationship.⁴ This would suggest that the actual tritium leakage rate might range from 0.015 to $0.10 \mu\text{Ci/s}$. Since the annual average noble radiogas leakage from a BWR is expected to be less than 0.1 Ci/s ($t = 30\text{m}$), the annual average tritium release rate from the fission source can be conservatively estimated at $0.12 \pm 0.12 \mu\text{Ci/s}$.

For the design basis total tritium appearance rate in the reactor and release rate in all plant effluents can be calculated from the sum of Equations 8-1 and 8-2 or, simply:

$$A_t = 1.3 \times 10^{-4} P + 0.12 \quad (8-3)$$

Tritium formed in the reactor is generally present as tritiated oxide (HTO) and to a lesser degree as tritiated gas (HT). Tritium concentration in the steam formed in the reactor will be the same as in the reactor water at any given time. This tritium concentration will also be present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents will also have this tritium concentration. Condensate storage receives treated water from the radioactive waste system and reject water from the condensate system. Thus, all plant process water will have a common tritium concentration.

Off-gases released from the plant will contain tritium, which is present as tritiated gas (HT) resulting from reactor water radiolysis as well as tritiated water vapor (HTO). In addition, water vapor from the turbine gland seal steam packing exhausters and a lesser amount present in ventilation air due to process steam leaks or evaporation from sumps, tanks and spills on floors will also contain tritium. The remainder of the tritium will leave the plant in liquid effluents or with solid wastes.

Recombination of radiolysis gases in the air ejector off-gas system will form water, which is condensed and returned to the main condenser. This tends to reduce the amount of tritium leaving in gaseous effluents. Reducing the gaseous tritium release will result in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged will also result in a higher process coolant equilibrium tritium concentration.

Essentially, all tritium entering the primary coolant will eventually be released to the environs, either as water vapor and gas to the atmosphere, or as liquid effluent to the plant discharge or as solid waste. Reduction due to radioactive decay is negligible due to the 12-y half-life of tritium.

The USPHS study⁴ at Dresden 1 estimated that approximately 90% of the tritium release was observed in liquid effluent, with the remaining 10% leaving as gaseous effluent. Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium will leave as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60 and 90% with the remainder leaving in gaseous effluent.

9. CONSIDERATIONS FOR FUTURE CHANGES TO SOURCE TERMS

In this report we have indicated how the 1971 design basis radioactive materials source terms were developed. In the period following the introduction of these updated source terms, further consideration has been given to completeness, clarity and updating based on additional operating experience. The influence of plant size and fuel power density is again considered. One limitation of the present source terms is the lack of definition on higher and lower source term values and their frequency of occurrence. This type of information is becoming available. In Figure 9-1, the leakage rate of Xe-133 per MWt is shown as a function of percent of operating time. Experimental data from KRB and Oyster Creek are included. The noble radiogas leakage rate seems to be plant size dependent; however, the radiohalogen leakage rates do not appear to be plant size dependent.

For purposes of plant performance evaluation, it is suggested that the operating average leakage rate value for Xe-133 of $3 \mu\text{Ci/s/MWt}$ be used; the remaining noble gas radioisotopes can be calculated from Equation 3-2 and the rated power of the BWR power plant being evaluated. If high levels of release for shorter time periods need to be evaluated, the leakage rate value for Xe-133 can be selected from the proposed design basis curve shown in Figure 9-1.

Similarly for the halogen radioisotopes, it is suggested that the operating average of $225 \mu\text{Ci/s}$ I-131 be used for purposes of plant performance evaluation. The remaining halogens can be calculated using Equation 4-1. If higher levels of release for shorter time periods need to be evaluated, the leakage rate value for I-131 can be selected from the proposed design basis curve shown in Figure 9-2.

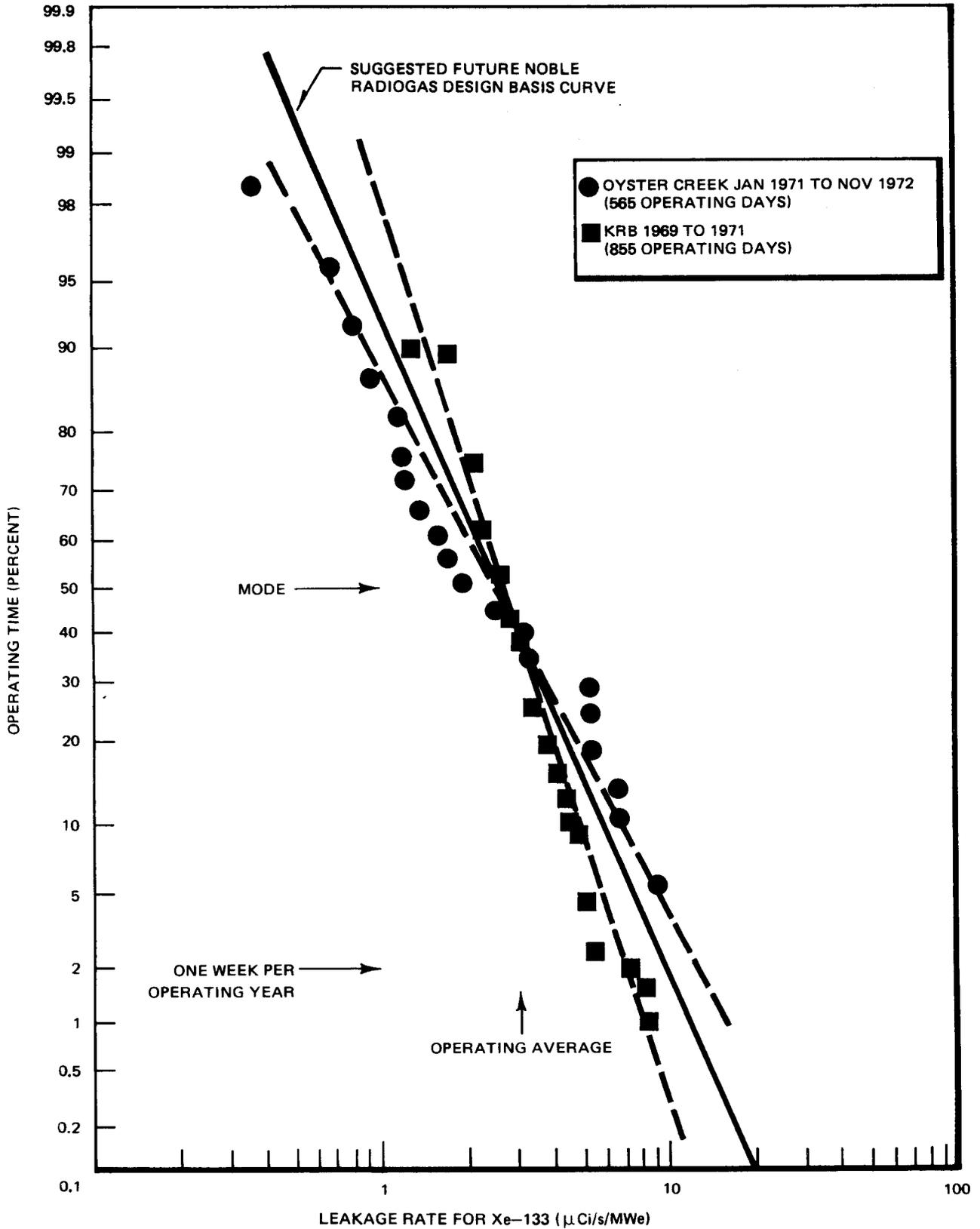


Figure 9-1. Xe-133 Leakage Rate vs Percent of Operating Time

10. NOMENCLATURE

- A_g = Leakage rate of a noble gas radioisotope ($\mu\text{Ci/s}$)
 A_h = Leakage rate of a halogen radioisotope ($\mu\text{Ci/s}$)
 A_r = Leakage rate of radioisotopes from a recoil source ($\mu\text{Ci/s}$)
 A_s = Leakage rate of nonvolatile soluble fission product radioisotope ($\mu\text{Ci/s}$)
 A_z = Leakage rate of an insoluble fission product radioisotope ($\mu\text{Ci/s}$)
 A_{at} = Formation rate of tritium by deuterium activation ($\mu\text{Ci/s/MWt}$)
 A_{dt} = Leakage rate of tritium from fuel ($\mu\text{Ci/s}$)
 b = Decay constant exponent (dimensionless)
 For noble radiogases: $b = 1 - m$
 For radiohalogens: $b = 1 - n$
 C_h = Concentration of a halogen radioisotope ($\mu\text{Ci/kg}$)
 C_s = Concentration of a nonvolatile soluble fission product radioisotope ($\mu\text{Ci/kg}$)
 C_z = Concentration of an insoluble fission product radioisotope ($\mu\text{Ci/kg}$)
 ϵ = Demineralizer removal fraction (dimensionless)
 f = Cleanup system flow (kg/s)
 F = Steam flow (kg/s)
 ϕ = Carryover fraction, ratio of concentration in steam or condensate to reactor coolant concentration (dimensionless)
 Φ = Thermal neutron flux ($n/\text{cm}^2\text{s}$)
 K_g = A constant establishing the magnitude of noble radiogas leakage
 K_h = A constant establishing the magnitude of radiohalogen leakage
 K_r = A constant establishing the magnitude of recoil leakage
 K_t = A constant establishing the magnitude of tritium leakage from fuel
 K'_g = $3.7 \times 10^4 K_g$
 K'_h = $3.7 \times 10^4 K_h$
 K'_r = $3.7 \times 10^4 K_r$

- λ = Decay constant of a radioisotope (s^{-1})
 m = Noble radiogas decay constant exponent (dimensionless)
 M = Mass of water in an operating reactor (kg)
 n = Radiohalogen decay constant exponent (dimensionless)
 P = Reactor rates power level (MWt)
 R_g = Source term for noble gas radioisotopes (fission/s)
 R_h = Source term for halogen radioisotopes (fission/s)
 R_r = Recoil source term (fissions/s)
 R_s = Source term for nonvolatile soluble fission product radioisotopes (fissions/s)
 R_z = Source term for insoluble fission product radioisotopes (fissions/s)
 Σ = Macroscopic thermal neutron cross section (cm^{-1})
 t = Decay following release from fuel (s)
 T = Fuel irradiation time (s)
 V = Coolant volume in core (cm^3)
 y = Fission yield of each fission product radioisotope (atoms/fission)

OTHER SYMBOLS AND ABBREVIATIONS

- s = seconds
 m = minutes
 h = hours
 d = days
 y = years
 Ci = curie
 μCi = 10^{-6} curies
 MWt = megawatts thermal
 cm = centimeter
 kg = kilogram

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